4 TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

In the LRA, Section 4.1, the applicant identifies the time-limited aging analyses (TLAAs) applicable to Arkansas Nuclear One, Unit No. 1 (ANO-1). The NRC staff reviewed the information in the license renewal application (LRA) to determine whether the applicant provides adequate information to meet the requirements stated in 10 CFR 54.21(c)(1).

4.1.1 Summary of Technical Information in the Application

In the LRA, Table 4.1-1, the applicant identifies the calculations and analyses that satisfied the six criteria of 10 CFR 54.3 for a TLAA. The applicant identifies the following as TLAAs:

RCS Piping

- metal fatigue
- analytical evaluation of flaws
- leak before break analysis
- thermal stratification

Pressurizer

- metal fatigue
- analytical evaluation of flaws

Reactor Vessel

- metal fatigue
- analytical evaluation of flaws
- intergranular separation
- thermal shock
- flow-induced vibration (FIV) analysis

Reactor Vessel Internals

- metal fatigue
- analytical evaluation of flaws
- FIV analysis
- stress and deflection analyses

Once-Through Steam Generators

- metal fatigue
- analytical evaluation of flaws

Reactor Coolant Pumps

- metal fatigue

- analytical evaluation of flaws

Control Rod Drive Mechanism Pressure Boundary

- metal fatigue
- analytical evaluation of flaws

Concrete Reactor Building Tendon

- loss of prestress

Reactor Building Liner Plate and Penetrations

- fatigue analysis

Spent Fuel Racks

- aging of Boraflex

Electrical Equipment

- environmental qualification

Reactor Coolant Pump Motor Flywheels

- fatigue crack growth

4.1.2 Staff Evaluation

In the LRA, Section 4.1, the applicant describes the requirements for identifying and evaluating TLAAs and plant-specific exemptions based on TLAAs. The applicant reviewed plant-specific documents including the ANO-1 licensing correspondence file, the ANO-1 updated final safety analysis report (UFSAR), Babcock and Wilcox (B&W) topical reports referenced in correspondence and in the UFSAR, and American Society of Mechanical Engineers (ASME) Section XI summary reports. The information provided by the applicant was reviewed by the NRC staff to determine which analyses and calculations met the six criteria defining TLAAs in 10 CFR 54.21(c)(1).

4.1.3 Conclusions

The NRC staff concludes that the applicant has provided a list of acceptable TLAAs as defined in 10 CFR 54.3, and that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA as defined in 10 CFR 54.3.

4.1.4 References for Section 4.1

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
4. NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," Revision 0, March 1996.

4.2 Reactor Vessel Neutron Embrittlement

The TLAAs for evaluating the effects of neutron irradiation on the ability of the reactor vessel to resist failure during a pressurized thermal shock (PTS) event, and the maintenance of acceptable Charpy upper shelf energy (USE) levels are discussed in Section 4.2 of the LRA.

4.2.1 Technical Information in the Application

The applicant participated in a Babcock and Wilcox Owners Group (B&WOG) effort that produced a series of topical reports to demonstrate that the aging effects for the reactor coolant system are adequately managed for the period of extended operation. One report, BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," addresses the reactor vessel and the applicable TLAAs. Staff review of this topical report and the applicant's responses to the review are given in Section 3.3.2.4 of this safety evaluation report (SER).

The TLAAs evaluated in the ANO-1 LRA include analyses and calculations performed to show compliance with 10 CFR 50.60, and 50.61, and Appendix G to 10 CFR Part 50, concerning PTS and acceptable Charpy USE levels. These are reviewed by the staff in the following paragraphs.

4.2.2 Staff Evaluation

BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," was reviewed and approved by the NRC staff in a letter dated April 26, 1999. In the LRA, Section 4.18, the applicant states that the ANO-1 reactor vessel integrity program is being utilized to ensure that the time dependent parameters used in the TLAA evaluations, as reported in BAW-2251A, are tracked such that the TLAA analyses remain valid for the period of extended operation. The staff reviewed the reactor vessel integrity program in Section 3.3.2.4.2.2 of this SER, and finds it acceptable for the period of extended operation. The two aspects of reactor vessel embrittlement are reactor vessel resistance to failure during PTS events and the maintenance of acceptable Charpy USE levels. Both are connected with neutron irradiation, for which the maximum anticipated effects would be in the reactor vessel beltline region at the end of the period of extended operation. A discussion of the two TLAAs is provided below.

Pressurized Thermal Shock

Rules for protecting against PTS in pressurized water reactors are given in 10 CFR 50.61(b)(1). Licensees are required to perform an assessment of the reactor vessel material's projected values of PTS reference temperature, RT_{PTS}, through the end of their operating license. With the potential approval of its application for an extended period of operation for ANO-1, this period will be through 48 EFPY.

In the LRA, Section 4.2.1, the applicant includes a description of the two options for determining RT_{PTS} for reactor vessel materials. As stated in 10 CFR 50.61(b)(1) the two acceptable methods for determining RT_{PTS} are as follows:

- Position 1 - for material that does not have surveillance data available

- Position 2 - for material that has surveillance data

Availability of surveillance data is not the only measure of whether Position 2 may be used. The data must also meet the credibility criteria given in the PTS rule (10 CFR 50.61).

Using the terminology in 10 CFR 50.61, RT_{PTS} is the sum of the initial (unirradiated) reference temperature, $RT_{NDT(u)}$, the shift in reference temperature caused by neutron irradiation (ΔRT_{NDT}), and a margin term (M) to account for uncertainties.

$RT_{NDT(u)}$ is determined using the method of Section III of the ASME Boiler and Pressure Vessel Code. That is, $RT_{NDT(u)}$ is the greater of the drop weight nil-ductility transition temperature or the temperature that is 15.6°C (60°F) below that at which the material exhibits Charpy test values of 50 ft-lbs and 35 mils lateral expansion. For a material for which test data are unavailable, generic values may be used if there are sufficient test results for that class of material. For Linde 1092, 0091, and 124, the generic value of $RT_{NDT(u)}$ is -48.9°C (-56°F). For Linde 80 weld material, with the exception of WF-70, the $RT_{NDT(u)}$ is taken to be the currently NRC-accepted values of -20.6°C (-5°F) or 21.7°C (-7°F). The value of -20.6°C (-5°F) or 21.7°C (-7°F) is the statistical mean value of Linde 80 welds tested by B&W as documented in topical reports BAW-2166 or BAW-1803, respectively. The ANO-1 reactor vessel does not contain any Linde 80 WF-70 weld material. For forgings and plate material, measured values are used where available. Where not available, a B&W generic value of -16.1°C (3°F) is used for forgings.

For Position 1 materials (surveillance data not available), ΔRT_{NDT} is defined as the product of the chemistry factor and the fluence factor. The chemistry factor is a function of the material's copper and nickel content expressed as weight percent. Although not explicitly discussed by the applicant, the "best estimate" copper and nickel contents will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values from weld deposits made using the same weld wire heat number as the limiting weld. For ANO-1, best estimate values were obtained from BAW-2251A. The value of the chemistry factor is directly obtained from tables in 10 CFR 50.61. The fluence factor is calculated using end-of-license peak fluence at the clad-to-base metal interface for the material's location. Fluence values were obtained by extrapolation to 48 EFPY from the current 32 EFPY values.

For Position 2 materials (surveillance data available), the discussion above for Position 1 applies except for determination of the chemistry factor, which in this instance is a material-specific value calculated as follows:

- multiply each ΔRT_{NDT} value by its corresponding fluence factor
- sum these products
- divide this sum by the sum of the squares of the fluence factors.

The applicant does not discuss the ratio procedure in 10 CFR 50.61. If surveillance data are being used and there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld (i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld), the measured values of ΔRT_{NDT}

must be adjusted for differences in copper and nickel by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld.

The margin term (M) is generally determined as follows:

$M = 2\,(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$
where σ_I is the standard deviation for $RT_{NDT(u)}$
and σ_Δ is the standard deviation for ΔRT_{NDT}

For determining M, $\sigma_I = 0$ if a measured value is used. If a generic value is used, σ_I is the standard deviation of the set of values used to obtain the mean value. For ΔRT_{NDT}, σ_Δ = -2.2°C (28°F) for welds and -8.3°C (17°F) for base metal (plate and forging), except that σ_Δ need not exceed one-half of the mean value of $\Delta RT_{NDT.}$ Note that when using Position 2, the same method for determining the σ values is used except that σ_Δ values may be halved (-10°C [14°F] for welds and -31.16°C [8.5°F] for base metal).

In accordance with 10 CFR 50.61(b)(2), the screening criteria for RT_{PTS} is 132.2°C (270°F) for plates, forgings, and axial welds, and 148.9°C (300°F) for circumferential welds. The values of RT_{PTS} at 48 EFPY for ANO-1 are given in Appendix A, Table A-1, of BAW-2251A. The RT_{PTS} values are shown to be below the screening criteria through 48 EFPY.

In a letter to the NRC dated July 1, 1998, the applicant submitted its response to an RAI regarding Supplement 1 to GL 92-01, Revision 1, "Reactor Vessel Structural Integrity." The information was also contained in the B&WOG topical report BAW-2325. In this response, the applicant states that after review of BAW-2325, the staff noted changes in the transition temperature shift data for certain surveillance capsules and issued several requests to the B&WOG for additional information. Subsequent interactions between the B&WOG and the staff resulted in the publication of Revision 1 to BAW-2325 in February 1999. Since BAW-2251A was completed prior to the BAW-2325, Revision 1, an assessment was performed by the applicant relative to the staff's findings regarding chemistry factors reported in BAW-2251A. The chemistry factors reported in BAW-2251A are equivalent to, or exceed, the chemistry factors reported in BAW-2325, Revision 1, for the limiting beltline welds at ANO-1. In addition, ANO-1 has recalculated the 48 EFPY fluence for the beltline region using the methodology described in BAW-2251A, Appendix D, and BAW-2241AP and has determined that the 48 EFPY fluence estimates reported in BAW-2251A remain conservative. Therefore, the 48 EFPY RT_{PTS} values for the limiting beltline welds reported in BAW-2251A, Table A-1, remain conservative for ANO-1 since both the chemistry and fluence estimates remain conservative.

In order to avoid exceeding the PTS screening criteria at ANO-1 during the period of extended operation, the applicant utilizes low leakage core designs. In addition, the applicant is involved with various industry activities that provide new information or new analysis techniques associated with the reactor vessel beltline region.

The limiting material for ANO-1 at the end of the license renewal period (48 EFPY) is projected to be weld WF-112 (weld wire heat number 406L44). The RT_{PTS} value was calculated using Position 1 in 10 CFR 50.61. The limiting projected RT_{PTS} value for ANO-1 is below the screening criterion at the end of the license renewal period. The limiting weld is the upper to lower shell circumferential weld with material identification WF-112 and weld wire heat number

406L44. It has a projected value of RT_{PTS} at 48 EFPY of 136.7°C (278°F) (the screening criterion is 148.9°C (300°F) for circumferential welds). Therefore, the staff found that, with respect to PTS events, the ANO-1 reactor vessel has sufficient margin to perform its intended function over the period of extended operation.

Charpy Upper-Shelf Energy

Although not discussed by the applicant, Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have Charpy USE levels in the transverse direction for the base metal and along the weld for the weld material according to the ASME Code, of no less than 102 J (75 ft. lbs.) initially, and must maintain Charpy USE levels throughout the life of the vessel of no less than 68 J (50 ft. lbs.). However, Charpy USE levels below these criteria may be acceptable if it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The B&WOG position on USE for 32 EFPY is documented in its responses to GL 92-01, Revision 1, "Reactor Vessel Structural Integrity" as reported in BAW-2166 and BAW-2222. The B&WOG position on USE for 48 EFPY is documented in BAW-2275, which is included in BAW-2251A as Appendix B.

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides two positions for determining Charpy upper-shelf energy (C_vUSE). Position 1 is for material that does not have surveillance data available and Position 2 is for material that does have surveillance data. For Position 1, the percent drop in C_vUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of RG 1.99, Revision 2. This percent drop is then applied to the initial C_vUSE to obtain the adjusted C_vUSE. For Position 2, the percent drop in C_vUSE is determined by plotting the available surveillance data on Figure 2 of RG 1.99, Revision 2 and fitting the data with a line drawn parallel to the existing lines that upper bounds all the plotted points. Again, the percent drop is determined, and used to adjust the initial C_v USE value.

Charpy USE issues are discussed in Section 4.2.2 of the application. The 48 EFPY C_v USE values determined for the ANO-1 reactor beltline materials are given in BAW-2251A, Table 4-4. The T/4 fluence values in this table were calculated in accordance with the ratio of the clad-to-base metal interface fluence to T/4 fluence values (i.e., neutron fluence lead factors at T/4) determined in the last reactor vessel surveillance program report. Table 4-4 shows that the C_vUSE is maintained above 68 J (50 ft. lbs.) for all base materials (plates and forgings), but weld materials nearly always fall below the 68 J (50 ft. lbs.) limit at 48 EFPY. Appendix G of 10 CFR Part 50 provides for this situation by allowing lower values of C_vUSE if it is demonstrated that the lower C_vUSE will provide margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. An equivalent margins analysis was performed for 48 EFPY, and the results reported in Appendix A to BAW-2251A for service levels A, B, C, and D. For service levels A and B, the results demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated. Again, the results showed that there is a sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code.

As mentioned earlier in this evaluation, the applicant submitted a response to an RAI for ANO-1 regarding Supplement 1 to GL-92-01, Revision 1. This response was BAW-2325, Revision 1. The "best estimate" chemistry composition (copper and nickel) was reported in BAW-2325, Revision 1. Best estimate chemistry compositions were also reported in BAW-2251A, and were summarized in Table A-1 of Appendix A to BAW-2251A for the various reactor vessel materials. The copper composition reported in BAW-2251A is equivalent to, or exceeds, the copper content reported in BAW-2325, Revision 1. In addition, the 48 EFPY fluence estimates were recalculated using the methodology described in Appendix B of BAW-2251A. It was shown that the fluence estimates listed in BAW-2251A remain conservative. Therefore the C_vUSE values, given in Table 4-4 of BAW-2251A, remain conservative.

The Appendix K analysis, from Section XI of the ASME Boiler and Pressure Vessel Code involves a quantitative assessment of the impact of low C_vUSE on reactor vessel integrity. In Appendix K analysis, cracks are postulated at the inner reactor vessel wall. Since the neutron fluence decreases with depth into the vessel, the Appendix K analysis method assumes the fracture toughness at the crack tip will be greater than that at the inner wall of the vessel. The applicant's analysis was carried out using conservative stress assumptions for service levels A, B, C, and D for 48 EFPY. The analysis, given in Appendix B of BAW-2251A, shows that for service levels A and B, there is sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated, and again the analytical results demonstrated that there is a sufficient margin beyond that required by Appendix K to Section XI of the ASME Code. The applicant concludes that evaluations for all four service levels show the adequacy of safety against fracture for the ANO-1 vessel for 48 EFPY.

The staff found the B&WOG evaluation of the Charpy USE acceptable for all ANO-1 materials for the period of extended operation because the 48 EFPY analysis reported in Appendix B of BAW-2251A, and referenced in this application, meets the provisions of 10 CFR 54.21(c)(1)(ii) and applies to ANO-1.

4.2.3 Conclusions

The staff has reviewed the TLAAs concerning irradiation-induced changes in reactor vessel material that affect PTS resistance and Charpy USE levels. On the basis of its review, the staff concludes that the applicant's PTS and USE analyses satisfy the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.4 References for Section 4.2

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One – Unit 1, License Renewal Application," January 31, 2000.

4. BAW-2251A, “Demonstration of the Management of Aging Effects for the Reactor Vessel,” B&WOG Generic License Renewal Program, June 1996.
5. NRC GL 92-01, Revision 1, Supplement 1, “Reactor Structural Integrity,” May 19, 1995.
6. BAW-2325, “Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity – Generic Letter 92-01, Revision 1, Supplement 1,” B&WOG, May 1998.
7. BAW-2325, “Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity, Revision 1, Supplement 1,” B&W Owners Group, January 1999.
8. BAW-2241AP, “Fluence and Uncertainty Methodologies,” April 1997.
9. BAW-2166, “Response to Generic Letter 92-01,” June 1992.
10. BAW-2222, “Response to Closure Letters to Generic Letter 92-01, Revision 1,” June 1994.
11. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.”
12. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components.”
13. 1CAN079801, Letter from D. James (ANO) to the NRC, “Generic Letter 92-01, Supplement 1, Reactor Vessel Structural Integrity, Request for Additional Information,” July 1, 1998.

4.3 Metal Fatigue

A metal component subjected to cyclic loads may fail at a load magnitude less than its ultimate load capacity due to metal fatigue, initiating and propagating cracks in the material. The fatigue life of a component is a function of its material, its environment, and the number and magnitude of the applied cyclic loads. Fatigue was a design consideration for piping and components in the ANO-1 RCS and, consequently, fatigue is part of the current licensing basis (CLB) for ANO-1. The applicant identifies fatigue and flaw growth evaluations as TLAAs for the piping and components of the RCS. The staff reviewed Section 4.3 of the LRA, which discusses thermal fatigue and flaw growth, as well as other related fatigue programs.

4.3.1 Technical Information in the Application

In the LRA, Section 4.3, the applicant discusses design criteria for thermal fatigue of RCS piping and components. The B&W scope of supply includes major components in the RCS, and the associated interconnecting piping. Vessels were designed in accordance with ASME Section III, 1965 edition, with addenda through the summer 1967. Reactor coolant pumps were designed in accordance with ASME Section III, 1968 edition. RCS piping supplied by B&W was designed to Nuclear Piping Code, USAS B31.7 Class 1. Bechtel-supplied piping includes the Class 1 portions of ancillary systems that are attached to the B&W scope of supply and miscellaneous vents, drains, and instrumentation lines. Bechtel-supplied piping was designed to Nuclear Piping Code USAS B31.7, Class 1, dated February 1968, and as corrected by Errata date of June 1968, or later appropriate ASME Section III Code sections, provided they have been reconciled.

The applicant's TLAA evaluation addresses the following topics:

- thermal fatigue, with separate consideration of environmentally-assisted fatigue, thermal stresses in piping connected to the RCS, and pressurizer line thermal stratification
- the ANO-1 transient cycle logging program
- flaw growth evaluation to demonstrate compliance with the ASME Section XI Inservice Inspection Requirements

4.3.2 Staff Evaluation

As discussed in the previous section, components of the RCS were designed to codes that contained explicit criteria for the fatigue analysis. Consequently, the applicant identifies the fatigue analyses and the flaw growth evaluations of the RCS components as TLAAs. The staff reviewed the applicant's evaluation of RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for RCS components involves calculating the cumulative usage factor (CUF). The fatigue damage caused by each thermal or pressure transient depends on the magnitude of the stresses caused in the component by a transient. The CUF sums the fatigue resulting from each transient. The applicant indicates that it addresses fatigue by ensuring that its effects are adequately managed for the period of extended operation.

For the B&W-supplied components, the design cyclic loadings and thermal conditions are defined by the component design specifications. The component design specification defines the transient cycle assumptions used in the fatigue evaluations for the component. As part of the B&W Generic License Renewal Program, the applicant was involved in a review to determine which Class 1 components were more sensitive to fatigue (environmentally-assisted fatigue was not considered), and which transients caused the greatest impact in terms of fatigue stress on the components. For this set of design transients, the number of transients accrued was compiled and a conservative projection was made to determine if the number of design transients would be exceeded in the period of extended operation. The applicant determines that, in no instance for ANO-1, did the extrapolation exceed the number of allowable design cycles prior to 60 years of operation.

For the Bechtel-supplied piping, the design cycle loading and thermal conditions are defined in a Bechtel Class 1 piping design specification. Existing cumulative usage factors and analyzed thermal transients documented in thermal fatigue calculations for the piping were reviewed by the applicant. On the basis of the number of transient cycles accrued for ANO-1 and the rate these cycles have been accumulated, the number of transient cycles that were originally projected for the current license term of 40 years envelopes the number of projected cycles to the end of a 60 year operating life.

The applicant has a process to log transient history and operating transient cycles. Applicable site procedures contain the responsibilities, logging requirements, reporting requirements and transient type definitions. Guidance is provided for collection of the necessary plant data and for projection of the number of cycles to end-of-life. The ANO-1 operating transient cycle logs are retained for the duration of the licence, per site procedures and ANO-1 TS. The applicant's procedure provides assurance that the number of plant transient cycles assumed in the design of the RCS components will not be exceeded.

Generic Safety Issue (GSI)-166, "Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of the RCS components. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," to address the period of extended operation for license renewal. The NRC closed GSI-190 in December 1999, concluding the following:

> *The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (NEI and EPRI), and the different approaches available to the licensees to mange the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed. This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe breaks as plants continue to operate. Thus, the staff concludes that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as AMPs are formulated in support of license renewal.*

From this NRC guidance on addressing environmental effects on fatigue, the applicant has adopted a procedure in which the specific locations in ANO-1 that are most susceptible to failure from thermal fatigue, and other degradation mechanisms, are determined by analysis. The calculations include consideration of stress level and lower bound material properties. From this information the applicant includes the most susceptible components in an augmented inservice inspection program.

The applicant lists the following critical component locations in B&W plants that are applicable to ANO-1:

- reactor vessel shell and lower head
- reactor vessel inlet and outlet nozzles
- pressurizer surge line
- makeup/high pressure injection nozzles
- reactor vessel core flood nozzle
- decay heat removal system Class 1 piping

The B&WOG conducted an environmentally-assisted fatigue analysis for the reactor vessel and documented it in BAW-2251A. This study derived environmental fatigue factors based on the model described in NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LRA Environments." These factors were applied to the reactor vessel shell and lower head, the vessel inlet and outlet nozzles, and the core flood nozzles. The study concluded that, after an accounting for environmentally-assisted fatigue, the reactor vessel fatigue usage factors remain acceptable for the period of extended operation. The applicant, therefore, concludes that reactor vessel shell, lower head, inlet and outlet nozzles, and core flood nozzles are no longer issues with respect to environmentally-assisted fatigue during the period of extended operation. In its April 26, 1999, SER for BAW-2251A, the NRC staff concluded that the environmental effects on the fatigue life of reactor vessel components had been adequately addressed for license renewal.

The applicant indicates that the three remaining locations (the pressurizer surge line, makeup/high pressure injection nozzles, and the decay heat removal system Class 1 piping) are included in the risk-informed ISI (RI-ISI) program which has recently been approved by the NRC staff as an alternative to requirements of ASME Section XI Inservice Inspection. The primary objective of the program was to identify "risk important" piping sections for inspection based on the analysis of the probability, and the consequences of piping failure. The applicant concludes that implementation of the RI-ISI program will ensure that inspections at ANO-1 will be performed in locations where degradation mechanisms, including thermal fatigue, are most likely to occur. In a letter to the applicant dated May 5, 2000, the staff requested additional information regarding the potential for fatigue cracking in the three remaining locations considering the data contained in NUREG/CR-5704, "Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steels."

In its response, the applicant outlines the environmentally-assisted fatigue analyses that were carried out for the three components specified by the staff. Specifically, the applicant states that the work to close out GSI-190 includes a review of the results of the INEEL studies published in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," as well as later experimental studies by ANL to account for the detrimental effects of primary coolant on the fatigue life. Using the environmental fatigue data in NUREG/CR-5704, the applicant's evaluation indicates that the surge line and the HPI/MU nozzles and safe ends have CUFs that may exceed 1.0 during the period of extended operation. For the decay heat removal piping, the CUF calculated from 1986 Code rules is less than 1.0. To address the locations where the CUF may exceed 1.0 when environmental effects are considered, the applicant proposed a program to manage the effects of fatigue. This program will be undertaken prior to the period of extended operation and will include one or more of the following options:

- refinement of the fatigue analysis in an attempt to lower the CUF to less than 1.0
- repair of affected locations
- replacement of affected locations
- management of the effects of fatigue during the period of extended operation using a program that will be reviewed and approved by the staff

The applicant commits to provide the NRC with the inspection details of the aging management program (AMP) requiring staff approval for managing the effects of fatigue prior to the period of extended operation if the last option is selected. As indicated by the applicant, the use of an AMP to manage fatigue will require prior staff review and approval. The staff found the applicant's proposed program an acceptable plant specific approach to address environmentally-assisted fatigue during the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1). However, in accordance with 10 CFR 54.21(d), this information needs to be added to the FSAR Supplement. This was FSAR Item 4.3.4 of Open Item 3.3-1.

The applicant also discusses actions taken in response to the NRC's Bulletin (BL) 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." In NRC BL 88-08, the staff requested that licensees review their RCS designs to identify any connected, non-isolable sections of pipe that could be subjected to temperature distributions that would result in unacceptable stresses. The applicant reviewed 23 different piping configurations connected to the RCS. As a result of its BL 88-08 reviews, the applicant added temperature monitoring devices to the HPI lines. The decay heat system suction line from the RCS also required monitoring and evaluation due to packing leaks on an isolation valve.

The applicant indicates that, because of the detection of stratified flow in ANO-2 lines, ANO-1 systems were reviewed again, to identify systems with attributes similar to the ANO-2 stratified lines. Four lines were found to require monitoring and evaluation. They are the pressurizer main spray, decay heat drop leg, RCS drains, and the RCS letdown drains. The applicant states that temperature monitoring and evaluation have demonstrated that these ANO-1 lines are qualified for their service conditions.

In response to BL 88-08, the applicant commits to perform enhanced ultrasonic examinations of 17 HPI welds, and visual inspection of two segments of HPI piping as part of its 10-year interval Inservice Inspection Plan. Subsequently, the scope of the ISI for the HPI lines and pressurizer surge line was modified based on an ANO-1 risk analysis performed consistent with the requirements of ASME Code case N-560, "Alterative Examination Requirements for Class 1, Category B-J Piping Welds, Section XI, Division 1." This commitment will be continued by the applicant through the period of extended operation.

In a letter to the applicant dated May 5, 2000, the applicant was asked to describe its modified inspection program for HPI welds and piping. In its response dated September 6, 2000, the applicant states that, initially, the inspections were to be ultrasonic for the welds and visual for the piping segments, in response to BL 88-08. As a result of the implementation of Code Case N-560, a new RI-ISI program was developed based on volumetric examination of the 13 most susceptible welds. Visual examination of the piping segments was eliminated from the program. The staff approved this modified program by a letter dated August 25, 1999.

The applicant discussed actions taken in response to NRC BL 88-11, "Pressurizer Surge Line Thermal Stratification." In BL 88-11, the staff requested that licensees establish and implement a program to confirm the integrity of the surge line, and to inform the NRC of actions taken to resolve the issue.

Originally, the applicant committed to performing enhanced ultrasonic examination of two elbows of the pressurizer surge line as part of the ANO-1 10-year interval ISI plan in response to BL 88-11. Subsequently, the scope of ISI inspections of the surge line was modified based on an ANO-1 risk analysis performed consistent with the requirements of ASME Code Case N-560, "Alternative Examination Requirements for Class 1, Category B-J Piping Welds, Section XI, Division 1." The applicant commits to continuing the examination through the period of extended operation.

In a letter to the applicant dated May 5, 2000, the staff requested clarification on possible modifications to the ISI procedure for the ultrasonic examination of the two elbows in the surge line as a result of the adoption of a new RI-ISI plan. In its response to the NRC dated September 6, 2000, the applicant states that the commitments made for the ISI for the elbows in the surge line in response to BL 88-11, had not changed as a result of the implementation of risk-informed evaluations required by Code Case N-560.

In the LRA, Section 4.3.4.4, the applicant discusses the actions taken in response to cracking of HPI/MU nozzle cracking in B&W plants, described in Information Notice 82-09, "Cracking in Piping of Makeup Coolant Lines at B&W Plants"; Generic Letter 85-20, "Resolution of Generic Issue 69: High-Pressure Injection/Makeup Nozzle Cracking in Babcock and Wilcox Plants," and NRC Information Notice 97-46, "Unisolable Crack in High-Pressure Injection Piping." On the basis of the recommendations by a B&WOG task force, actions taken by the applicant include repair of nozzles with loose or damaged sleeves, maintenance of adequate minimum flow, implementation of augmented inspection programs for the nozzles, and performance of stress analysis with modified thermal sleeves. The augmented ISI program for the HPI/MU nozzles is consistent with the methodology and scope of inspection recommended by the B&WOG Safe-End Task Force. The applicant commits to ultrasonic testing of the knuckle region of the HPI nozzles every fifth refueling cycle, and radiography of the thermal sleeves will continue through the period of extended operation. In a letter to the NRC dated May 5, 2000, the applicant states that there will be radiographic testing of the sleeves and the gap between the safe-ends and the sleeves every fifth refueling cycle to monitor for cracking during the period of extended operation. The staff agrees that the augmented inspection program provides an acceptable method to manage cracking of the HPI/MU nozzles during the period of extended operation.

In the LRA, Section 4.3.6, the applicant discusses flaw growth evaluation. The applicant states that indications detected during ISI that exceed that specified acceptance criterion could be analytically evaluated using crack growth analysis. The crack growth analyses would consider the same design transient cycle assumptions used in the original design. Since the analyses were performed using the full number of design transient cycles, which have been demonstrated to be applicable over 60 years of operation, these flaw growth calculations remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The applicant uses ASME Code Case N-481 to evaluate RCP weld flaws. The Code Case N-481 flaw tolerance evaluation was reviewed by the applicant to determine if the evaluation is acceptable for the period of extended operation. A separate effort was carried out to evaluate the acceptability of a Code Case N-481 flaw growth analysis for the RCS pump casings for the period of extended operation, taking into consideration the effects of thermal aging on fracture toughness. The fatigue growth calculation performed, included an assumption of 240 heatup and cooldown cycles. Since the applicant has not increased the number of design transients for license renewal, the applicant states that the flaw growth evaluation for pump casings is acceptable for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The staff found that the applicant's TLAA evaluation meets the requirements of 10 CFR 54.21(c)(1).

After its initial review, the staff requested that the FSAR Supplement include a summary description of the applicant's evaluation of NUREG/CR-6260 components for environmental fatigue including the options for future evaluations of the surge line and HPI/MU nozzles and safe ends in accordance with 10 CFR 54.21(d). This was FSAR Item 4.3.4 for Open Item 3.3-1.

In its revised summary description of the FSAR Supplement, Section 16.3.2, the applicant provides a description proposed program as described above to address environmental effects of fatigue including the options for future evaluations that meets the requirements of 10 CFR 54.21(d). The staff found the revised summary description submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

4.3.3 Conclusions

On the basis of its projection of the number of expected transients the applicant concludes that the fatigue analysis of RCS components and the flaw growth evaluation of indications found during component inspections will remain valid for the period of extended operation. The applicant also has a process to maintain a record of these transients and that process will continue during the period of extended operation. In addition, the applicant commits to implement a program, prior to the period of extended operation, to manage two locations in the RCS where environmental effects on the fatigue life are significant. On the basis of the applicant's TLAA evaluations, and its commitment to implement a program to manage the environmental effects on fatigue at critical locations prior to the period of extended operation, the staff concludes that the applicant's actions and commitments satisfy the requirements of 10 CFR 54.21(c)(1).

4.3.4 References for Section 4.3

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
4. ASME Boiler and pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components."
5. USAS B31.7, "Nuclear Power Piping."
6. NUREG/CR-6335, "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LRA Environments," August 1995.
7. NUREG/CR-5704, "Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.
8. NRC BL 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," June 22, 1988.
9. NRC BL 88-11, "Pressurizer Surge Line Thermal Stratification," December 20, 1988.
10. NRC IN 82-09, "Cracking in Piping of Makeup Coolant Lines at B&W Plants," March 31, 1982.
11. NRC GL 85-20, "Resolution of Generic Issue 69: High Pressure Injection/Makeup Nozzle Cracking in Babcock and Wilcox Plants," November 11, 1985.
12. NRC IN 97-46, "Unisolable Crack in High-Pressure Injection Piping," July 9, 1997.

4.4 Environmental Qualification

The ANO-1 10 CFR 50.49 Environmental Qualification (EQ) Program has been identified as a TLAA for the purposes of license renewal. The TLAA of EQ components includes all long-lived, passive and active electrical components and commodities located in a harsh environment that are important to safety, including safety-related and Q-list equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and the necessary post-accident monitoring equipment.

The NRC staff has reviewed Section 4.4, "Environmental Qualification," of the LRA to determine whether the applicant submitted adequate information to demonstrate that they meet the requirements in 10 CFR 54.21(c)(1) regarding an evaluation of the EQ TLAA. In addition, the staff reviewed Section 4.4.69, "GSI-168 'EQ of Electrical Components'."

On the basis of this review, the NRC staff requested additional information in letters to the applicant dated April 17, 2000, and April 25, 2000. The applicant responded to these RAIs in a letter to the NRC dated July 6, 2000. In addition, the NRC staff met with the applicant on May 25, 2000, to review related EQ calculations. The results of this meeting are documented in a letter from the NRC to the applicant dated June 13, 2000.

4.4.1 Technical Information in the Application

In the LRA, Section 4.4, the applicant describes the TLAA evaluation methodology and how the results from these evaluations were used to demonstrate that (i) the analyses remain valid for the period of extended operation; (ii) the analyses have been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The following is a summary of the methodology used by the applicant to evaluate the EQ TLAAs and the results from this evaluation.

Scope of EQ Equipment

The qualification requirements for electrical equipment originally installed at ANO-1 are based on NRC IE Bulletin 79-01B, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," which is now referred to as the Division of Operating Reactors (DOR) Guidelines. The applicant's EQ program complies with the scope of 10 CFR 50.49 requirements, and was "grandfathered" by 10 CFR 50.49 thereby allowing qualification in accordance with the DOR Guidelines. Therefore, the DOR Guidelines document is the CLB for the ANO-1 EQ program.

The Environmental Qualification Program at ANO-1 is a centralized plant support program administered by design engineering in order to maintain compliance with 10 CFR 50.49. The scope of the EQ program includes the following categories of electrical equipment located in a harsh environment:

- safety-related equipment
- non-safety-related equipment whose failure could adversely affect safety-related equipment

- the necessary post-accident monitoring equipment

The identification of EQ equipment is specified by procedural controls, and a component database is utilized to maintain an EQ equipment master list.

EQ Process

The EQ Program includes three main elements:

- establish and control a list of equipment and service conditions
- establish and control equipment documentation
- maintain (or preserve) qualification through preventive maintenance, the procurement process and corrective actions

As part of the first element, the applicant has established, and currently controls, an EQ master list of equipment, and the service condition for the harsh environment plant areas. The applicant has also established, and currently controls, the qualification documents, including vendor test reports, vendor correspondence, calculations, evaluations of equipment tested conditions as compared to plant required conditions, and determinations of configuration and maintenance requirements. Finally, the applicant established the following required processes to maintain the qualification:

- a preventive maintenance process for replacing parts and the equipment at required intervals
- a design control process to ensure that changes to the plant are evaluated to assess the potential impact on the EQ program
- a procurement process to ensure new and replacement equipment is purchased in accordance with applicable EQ requirements
- a corrective action process to identify and correct problems

Replacement of Equipment

As a normal part of the ANO-1 EQ process, when the EQ documentation process establishes that equipment, or parts thereof, have a limited life, the preventive maintenance process ensures that the equipment or parts are replaced before the expiration of the qualified life. The ANO-1 EQ program ensures that replacement equipment is purchased in accordance with applicable EQ requirements.

Reanalysis of the Qualified Life

If excess conservatism exists in the original qualified life determination, then reanalysis could be performed to extend the qualified life. The reanalysis would then become a part of the EQ documentation. Parameter conservatism may exist in the ambient temperature of the equipment, in an unrealistically low activation energy, and in the application of the equipment. The primary method used for reanalysis is to reduce excess conservatism in the equipment service temperature by using temperature values closer to an actual temperature measured in the area around the equipment being analyzed. This reanalysis is performed as follows:

- Analytical Methods - This reanalysis method uses standard EQ techniques, such as the Arrhenius methodology for thermal aging effects. Moisture has not been identified as a significant aging mechanism for ANO-1. The analytical method used for radiation analysis is to identify the 40-year radiation dose for the area where the equipment is installed, multiply that value by the ratio of the evaluation period divided by 40 years (i.e., 60 years/40 years = 1.5), and add the applicable accident radiation dose to obtain the total integrated dose for the equipment.

- Data Collection and Reduction Methods - The primary method used for reanalysis is to reduce excess conservatism in the equipment service temperatures. The applicant describes the following activities used to obtain temperature data for the reanalysis of EQ equipment:

  - A plant modification installed a temporary temperature monitoring system for the ANO-1 reactor building, and data were collected from 1989 to 1996. This system included temperature elements that monitored 21 different area ambient temperatures (at various elevations and azimuths) and 11 different EQ equipment surface temperatures.

  - In May 1989, the applicant conducted EQ walkdowns to determine the EQ equipment surface temperatures in the auxiliary building, and the temperatures in the associated general area.

  - Self-contained temperature data loggers were initially installed in the ANO-1 reactor building in 1993, and were used to gather additional temperature data.

  - In August/September 1997, the applicant conducted an environmental walkdown and documented area temperatures and any hot spots in several different buildings, including the auxiliary building. These measurements were taken at a single point in time with a hand-held digital thermometer or infrared camera.

  - For the reactor building, the applicant measured temperature on, next to, or within close proximity to the EQ equipment. Measuring devices were located in expected hot areas, such as the D-rings of the containment. Measurements were taken with the temporary monitoring system on most normal working days, and data logger measurements were taken continuously. From the data obtained, the applicant determined the overall operating temperature, which is generally several degrees above the average temperature value, and is visually selected so that most data points fall on or below this value.

  - For the auxiliary building, the applicant measured temperatures on equipment surfaces and in the areas of the auxiliary building containing EQ equipment at a single point in time. To provide conservative values of the operating temperatures, the applicant took these measurements while the plant was operating during warm months of the year. An infrared camera was used to specifically identify hot spots.

- Underlying Assumptions - ANO-1 was one of several plants cited in NRC Information Notice 89-30 as having experienced elevated temperatures in the plant. For ANO-1, the reactor building experienced the elevated temperatures. This event was identified and the conditions were evaluated in 1987, including revising EQ analyses. In an effort to reduce the reactor building temperatures, the applicant installed larger chilled water pumps, an additional chiller, and an additional air-handling unit. These major plant modifications reduced the reactor building operating temperatures. Plant modifications or initiatives are controlled by procedures that include determining the related impact on EQ analyses. There have been no major plant modifications or events at ANO-1 that have changed the radiation values used in the EQ analyses.

Refurbishment of EQ Electrical Equipment

Refurbishment is an option at ANO-1. EQ equipment that is in need of refurbishment is refurbished in place or is replaced with new equipment or previously refurbished equipment taken out of storage before exceeding its qualified life. Refurbishment is a process that preserves the qualification status of equipment and is typically accomplished by replacing items such as gaskets, seals, and wires that are the limiting components or sub-components for the qualified life. The EQ documentation identifies limited-life replacement parts for specific equipment, manufacturers, and models. The replacement option discussed for several equipment types would effectively involve refurbishment.

Ongoing Qualification/Retesting

For EQ equipment with a qualified life less than the required design life of the plant, "ongoing qualification" is a method of long-term qualification involving additional testing. Ongoing qualification or retesting, as described in IEEE Standard 323-1974, Section 6.6, "Ongoing Qualification," is not currently considered by the applicant to be a viable option, and there are no plans to implement such an option. If this option becomes viable in the future, the applicant would perform ongoing qualification or retesting in accordance with accepted industry and regulatory standards.

Procurement of EQ Equipment

The ANO-1 EQ program includes procurement processes to ensure that new and replacement equipment is purchased in accordance with applicable EQ requirements.

Plant Environmental Changes

Controls used to monitor changes in plant environmental conditions involve temperature monitoring in the reactor building. For areas of the auxiliary building, the applicant relies upon normal operator rounds, personnel performing routine maintenance work, and periodic engineering walkdowns and inspections to identify changes in normal operating temperature conditions that might exceed the design value of 105°F.

EQ Generic Safety Issue (GSI)

To resolve GSI-168, "Environmental Qualification of Electrical Components," the applicant has chosen to submit a technical rationale demonstrating that the effects of aging will be managed in accordance with 10 CFR 54.21(c)(1)(iii) until some future point in time when other more reasonable options become available.

4.4.2 Staff Evaluation

The NRC staff reviewed Section 4.4 of the ANO-1 LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1). In addition, the NRC staff met with the applicant to obtain clarifications, to review specific EQ calculations, and to review the applicant's response to RAIs.

The NRC staff verified that the applicant is using standard, approved EQ methodologies and acceptance criteria as defined by NRC IE Bulletin 79-01B (DOR Guidelines), including Supplements 1, 2, and 3; NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1; 10 CFR 50.49, "Environmental Qualification for Electric Equipment Important to Safety for Nuclear Power Plants"; Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 1; and various EQ-related NRC generic letters, information notices; and SERs. The current ANO-1 actions, such as refurbish and replace, for short-lived EQ equipment are also acceptable for long-lived EQ equipment.

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(i)

In the LRA, Section 4.4.18, "General Atomic Radiation Detectors," the applicant states that it uses 10 CFR 54.21(c)(1)(i) in its TLAA evaluation to demonstrate that the analyses remain valid for the period of extended operation. The applicant applies this method to the detector assemblies, including their connectors, which are constructed of metal, ceramic, quartz cloth insulation, and Rexolite. With the exception of Rexolite, these materials are inorganic, and are not susceptible to thermal or radiation age degradation. The Rexolite is used in the connectors as a locator during assembly, and has no required function after assembly. On the basis of the NRC staff's review of the information submitted by the applicant regarding the General Atomic radiation detectors, and its materials of construction, the NRC staff finds the applicant's demonstration to be consistent with 10 CFR 54.21 (c)(1)(i).

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(ii)

For the following list of electrical equipment identified in Section 4.4 of the LRA, the applicant uses 10 CFR 54.21(c)(1)(ii) in its TLAA evaluation to demonstrate that the analyses have been projected to the end of the period of extended operation:

- 4.4.1 Allis Chalmers Motors
- 4.4.2 Anaconda Instrumentation Cable, FR-EP Insulation

- 4.4.3 Anaconda Control and Power Cable, EP Insulation
- 4.4.4 Anaconda EPR Insulated Instrumentation, Control/Power Cable
- 4.4.8 Buchanan Terminal Blocks, Outside Reactor Building
- 4.4.9 Buchanan Terminal Blocks, Inside Reactor Building
- 4.4.10 Conax Thermocouples
- 4.4.11 Conax Resistance Temperature Detectors
- 4.4.12 Conax Multipin Connector
- 4.4.13 Conax Electrical Penetration Assemblies
- 4.4.14 Conax Electrical Connection Seal Assembly
- 4.4.15 Conax Electrical Feedthrough Adapters
- 4.4.16 Eaton Flame Retardant Ethylene Propylene Diene Monomer Insulated Cable
- 4.4.17 Gems De Laval Level Sensors
- 4.4.19 General Electric Terminal Blocks
- 4.4.21 Limitorque Motor-Operated Valve Actuators; Alternating Current/Inside Reactor Building (most applications)
- 4.4.22 Limitorque Motor-Operated Valve Actuators; Alternating Current/Outside Reactor Building
- 4.4.23 Limitorque Motor-Operated Valve Actuators; Direct Current/Outside Reactor Building
- 4.4.26 NAMCO EA-740 Limit Switches with NAMCO Connectors
- 4.4.27 NAMCO EA-740 Limit Switches
- 4.4.28 NAMCO Quick Connectors
- 4.4.29 Okonite 5 kV Power Cable with EPR Insulation and an Okolon Jacket
- 4.4.30 Okonite 2 kV Power and Control Cable with Okonite or Okoguard Insulation and Okoprene or Okolon Jackets
- 4.4.31 Okonite 600V Power Cable with Okonite Insulation and an Okolon Jacket (most applications)
- 4.4.32 Okonite 600V Power Cable with FMR Insulation (most applications)
- 4.4.33 Okonite T-95 and No. 35 Splicing Tapes (most applications)
- 4.4.34 Raychem 600V Flamtrol XLPE Cable
- 4.4.35 Raychem Cable Splice and Jacket Repair Tape (type NJRT)
- 4.4.36 Raychem Cable Splices (types WCSF-N, NPK, NMCK, ANK, etc.)
- 4.4.37 Reliance Electric, Electric Motors
- 4.4.38 Rockbestos Coaxial Cable (most applications)
- 4.4.39 Rockbestos Firewall III Irradiation Cross-Linked Polyethylene Cable
- 4.4.40 Rockbestos Firezone R Silicone Rubber High-Temperature Cable (some applications)
- 4.4.41 Rockbestos Firewall III Chemically Cross-Linked Polyethylene Cable
- 4.4.44 Rotork Motor Operated Valve Actuators, Model NA1
- 4.4.45 Target Rock Solenoid-Operated Valves (Report 2375)
- 4.4.47 Target Rock Modulating Solenoid Operated Valves (Report 3414)
- 4.4.48 Target Rock Solenoid-Operated Valves (Reports 2375 and 1827)
- 4.4.49 TEC Valve Flow Monitoring System (some subcomponents)
- 4.4.50 TEC Reactor Vessel Level Monitoring System (some subcomponents)
- 4.4.51 Weed Resistance Temperature Detectors
- 4.4.52 Dow-Corning 3145 Silicone Sealant
- 4.4.55 Westinghouse Motors, Models TBFC and SBDP

- 4.4.56 Babcock & Wilcox Core Exit Thermocouples (pin half connector with mineral-insulated cable)
- 4.4.57 Gamma Metrics Neutron Detectors and Cable Assemblies (organic cable)
- 4.4.58 Brand Rex Cross-Linked Polyethylene Coaxial Cable
- 4.4.59 Brand Rex Cross-Linked Polyethylene Power and Control Cable
- 4.4.60 NDT International Acoustic Sensor, Connector and Cable
- 4.4.61 American Insulated Wire 600V Instrumentation Cable
- 4.4.62 American Insulated Wire 600V Power and Control Cable
- 4.4.63 AMP Pre-insulated Butt Splices
- 4.4.64 EGS Quick Disconnect Electrical Connectors (except connector o-rings)
- 4.4.65 EGS Grayboot Electrical Connectors
- 4.4.67 Valcor Model V526-5961-1 Solenoid Operated Valve
- 4.4.68 General Cable Corporation 5 kV Power Cable

In its response to the NRC's RAIs, the applicant supplied the following clarifications regarding two of the above components:

- Target Rock Solenoid-Operated Valves - the EQ documentation shows that the valves are qualified for more than 60 years at 82.2°C (180°F). Most applications are at or below 82.2°C (180°F). Applications at temperatures above 82.2°C (180°F) have been evaluated separately, and replacements are identified as a normal part of the ANO-1 EQ process. The applicant concludes that EQ aging analyses of Target Rock solenoid-operated valve have been projected to the end of the period of extended operation for most applications, and the remainder have scheduled replacements before they exceed the qualified life. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

- Westinghouse Motors - The EQ documentation shows that these motors are qualified for 78,840 hours of operation at their maximum operating temperature of 120°C (248°F). During plant operation, the motors do not run (i.e., they only run during surveillance). Between the surveillance and a conservative 1-year post-accident operating time (8,760 hours), a conservative total run-time of 18,000 hours is estimated. The applicant concludes that this is significantly less than the 78,840 hours for which these motors are qualified, and would conservatively consider them qualified (accounting for runtime and non runtime) through the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

On the basis of the review for the thermal and radiation summaries for the electrical equipment discussed above, the review of EQ calculations, and the responses to the NRC's RAI, the NRC staff found that the applicant has demonstrated that the analyses have been projected to the end of the period of extended operation consistent with 10 CFR 54.21(c)(1)(ii)

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(iii)

For the following list of electrical equipment identified in Section 4.4 of the LRA, the applicant uses 10 CFR 54.21(c)(1)(iii) in its TLAA evaluation to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation:

- 4.4.5 ASCO Solenoid Valves, Outside Reactor Building (some applications)
- 4.4.6 ASCO Solenoid Valves, Inside Reactor Building
- 4.4.7 Boston Insulated Wire, Instrumentation, Control, and Power Cable
- 4.4.20 ITT/General Controls Electro-Hydraulic Actuators
- 4.4.21 Limitorque Motor-Operated Valve Actuators; Alternating Current/Inside Reactor Building (some applications)
- 4.4.24 NAMCO EA-170 Limit Switches
- 4.4.25 NAMCO EA-180 Limit Switches
- 4.4.26 NAMCO EA-740 Limit Switches with NAMCO Connectors
- 4.4.31 Okonite 600V Power Cable with Okonite Insulation and an Okolon Jacket (some applications)
- 4.4.32 Okonite 600V Power Cable with FMR Insulation (some applications)
- 4.4.33 Okonite T-95 and No. 35 Splicing Tapes (some applications)
- 4.4.38 Rockbestos Coaxial Cable (some applications)
- 4.4.40 Rockbestos Firezone R Silicone Rubber High-Temperature Cable (some applications)
- 4.4.42 Rosemount Model 1153 Series D Pressure Transmitters
- 4.4.43 Rosemount Model 1154 Pressure Transmitters
- 4.4.46 Target Rock Solenoid-Operated Valves (Reports 2375 and 3996)
- 4.4.49 TEC Valve Flow Monitoring System (some subcomponents)
- 4.4.50 TEC Reactor Vessel Level Monitoring System (some subcomponents)
- 4.4.53 Westinghouse Hydrogen Recombiners
- 4.4.54 Westinghouse Motors, Model ABDP
- 4.4.56 Babcock & Wilcox Core Exit Thermocouple (except the pin half connector with the mineral-insulated cable)
- 4.4.57 Gamma Metrics Neutron Detectors and Cable Assemblies (except organic cable)
- 4.4.64 EGS Quick Disconnect Electrical Connectors (connector o-rings)
- 4.4.66 Valcor Model V526-5683 Solenoid Operated Valve

The NAMCO EA-740 Limit Switches with NAMCO connectors are replacement components for equipment removed from service in 1986. These replacements have a qualified life of 47.1 years at 40.6°C (105°F), and their qualified life expires in 2033, which is 1 year before the end of the period of extended operation. The applicant will replace this equipment in accordance with the ANO-1 EQ program before the end of the qualified life, unless an analysis is performed to extend the qualified life.

The Gamma Metrics Neutron Detectors and Cable Assemblies are not original plant equipment; they were installed in 1984 and are qualified in accordance with Regulatory Guide 1.97. The detector assemblies and junction box o-rings have a qualified life of 40 years at 48.9°C (120°F), and their qualified life expires in 2024, which is 10 years before the end of the period of extended operation. The applicant will replace this equipment in accordance with the ANO-1 EQ program before the end of the qualified life, unless an analysis is performed to extend the qualified life. Mineral insulated cable extending from the detector is non-age-sensitive. The organic cable is qualified for 50 years at 82.2°C (180°F), and its qualified life expires in 2034, which makes these cables qualified through the period of extended operation because the application is below 82.2°C (180°F).

The EGS Quick Disconnect Electrical Connectors in two applications at or below 65.6°C (150°F) are not original plant equipment; they were installed in 1995. These connectors have a qualified life of 40 years at 65.6°C (150°F), and their qualified life expires in 2035, therefore, they are qualified to the end of the period of extended operation. The connector o-rings are qualified for 10 years at 65.6°C (150°F), and their qualified life expires in 2005, which is 29 years before the end of the period of extended operation. The applicant will replace this equipment in accordance with the ANO-1 EQ program before the end of the qualified life, unless an analysis is performed to extend the qualified life.

The remaining components are original plant equipment with a qualified life of 40 years or less. In a response to the NRC staff's RAI, the applicant addressed the options of replacement, refurbishment or reanalysis for the above components. The applicant has no current plans to reanalyze and extend the qualified life of this equipment and will replace or refurbish the equipment before its qualified life expires, in accordance with the ANO-1 EQ program.

In cases where replacement is the current option, the applicant states that replacement requirements are managed through the preventive maintenance program in which a work package is automatically initiated and implemented to perform and document the replacement before exceeding the qualified life. Additionally, Westinghouse motors, Model ABDP are qualified in accordance with NRC IE Bulletin 79-01B requirements and, therefore, the components would be upgraded if they were to be replaced. If reanalysis is performed for this equipment, it would follow the same process discussed in Section 4.4.2 under "Reanalysis of the Qualified Life." The applicant has not yet decided which option will be used for this equipment.

The applicant did not identify any specific cases where refurbishment is the current option. However, in a response to the NRC's RAI, the applicant states that the replacement option discussed for several equipment types would effectively involve refurbishment. The staff found this acceptable because it is consistent with 10 CFR 50.49 and 54.21 (c)(1)(iii).

4.4.3 Conclusions

On the basis of the review described above, the NRC staff has determined that there is reasonable assurance that the applicant has evaluated the TLAAs for EQ of electrical equipment in accordance with 10 CFR 54.21(c)(1).

4.4.4 References for Section 4.4

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.
4. IEEE Std. 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," 1974.
5. C. I. Grimes letter to D. Walters (NEI), "Guidance on Addressing GSI 168 for License Renewal," Project 690, dated June 2, 1998.

6. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."
7. 10 CFR 50.49, "Environmental Qualification of Electric Equipment to Safety for Nuclear Power Plants."
8. 10 CFR 50.34(a)(1), "Contents of application; Technical Information."
9. 10 CFR Part 100, "Reactor Site Criteria."
10. NRC BL 79-01B, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors."
11. NRC IN 89-30 and IN 89-30, Supplement 1, "High-Temperature Environments at Nuclear Power Plants."
12. NRC IN93-39, "Radiation Beams from Power Reactor Biological Shields."

4.5 Concrete Reactor Building Tendon Prestress

The applicant identifies loss of reactor building prestress as a TLAA in the LRA. This section of the report documents the staff's safety evaluation of the TLAA for the reactor building tendon prestress based on information presented in Section 4.5 of the LRA.

4.5.1 Technical Information in the Application

In the LRA, Section 4.5, "Concrete Reactor Building Tendon Prestress," the applicant presents the results of the TLAA for the loss of prestress in the post-tensioning system. The applicant states that the ANO-1 reactor building post-tensioning system is designed in accordance with ACI 318-63 for prestress losses caused by:

- seating anchorage
- elastic shortening of concrete
- creep of concrete
- relaxation of prestressed steel
- frictional loss due to curvature in the tendons and contact with tendon conduit

At the time of initial licensing, the initial stress from tensile loading and the appropriate prestress loss parameters were used by the applicant to calculate the design losses and the final effective prestress at the end of 40 years for the dome, vertical, and hoop tendons. In the LRA, the applicant states that this analysis is described in the ANO-1 UFSAR, Section 5.2.4.2.1, and identifies it as a TLAA requiring review for license renewal.

The applicant describes the requirements in ASME Code Section XI, Subsection IWL for the inservice inspection, repair, and replacement activities of the post-tensioning components of concrete containments, and identifies that tendon force and elongation measurements are required to evaluate the prestress forces in the post-tensioning system.

The applicant states that "ANO-1 is completing a calculation of the final effective tendon prestress based on additional information on concrete creep from existing creep tests and results of the tendon surveillance testing." The applicant indicates that the calculation will confirm projections on the relaxation of the tendons and this will show that the tendons will be acceptable for the period of extended operation.

The applicant also indicates that the ASME Section XI Inservice Inspection Program, IWL Inspections, will be adequate to manage the effects of aging on the intended function for the period of extended operation. The applicant states that the "implementation of this program dispositions this TLAA in accordance with 10 CFR 54.21(c)(1)(iii)."

4.5.2 Staff Evaluation

From the description provided in the LRA, Section 4.5, the applicant is currently performing a calculation that will confirm projections on the relaxation of the tendons and this will show that the tendons will be acceptable for the period of extended operation. This type of analysis would be consistent with a TLAA performed in accordance with 10 CFR 54.21(c)(1)(ii). However, after describing the ASME Section XI Inservice Inspection Program, IWL Inspections, the applicant states that "this program dispositions this TLAA in accordance with 10 CFR 54.21(c)(1)(iii)." Therefore, it is not clear which approach is being taken to address the TLAA for loss of tendon prestress.

A TLAA performed in accordance with 10 CFR 54.21(c)(1)(iii) must demonstrate that the effects of aging on the intended functions will be adequately managed for the period of extended operation. The information contained in LRA Section 4.5 is not sufficient for the staff to conclude that the loss of prestress will be adequately managed for the period of extended operation. If the applicant is addressing this TLAA in accordance with 10 CFR 54.21(c)(1)(iii), a description of the attributes of the AMP is needed, with special emphasis on parameters monitored, monitoring and trending, acceptance criteria, corrective actions, and operating experience.

In the LRA, Section 4.5, the applicant indicates that the analysis for prestress losses and final effective prestress at the end of 40 years is summarized in the ANO-1 UFSAR, Section 5.2.4.2.1. The NRC staff reviewed Section 5.2.4.2.1 as well as Section 5.2.4 of the ANO-1 UFSAR (Amendment 15), and could not locate the applicable information. Therefore, the staff requested that the applicant identify the section in the UFSAR that contains a description of the tendon prestress calculations corresponding to the end of 40 years. In order to understand the applicant's approach to address the TLAA for reactor building tendon prestress, the staff requested additional technical information in a letter to the applicant dated May 5, 2000.

In its response to the NRC dated September 7, 2000, the applicant states that the TLAA for loss of tendon prestress has been addressed in accordance with 10 CFR 54.21(c)(1)(iii). However, the additional information provided in the September 7, 2000, letter did not adequately address monitoring and trending, acceptance criteria, and corrective action.

For the purpose of monitoring and trending of tendon prestress, the following parameters need to be plotted against time and projected for the period of extended operation: the predicted lower limit (PLL), the minimum required value (MRV), and the trend line representing the measured prestress forces. 10 CFR 50.55a(b)(2)(viii)(B) specifies acceptance criteria for trending of tendon forces, in addition to the criteria contained in ASME Section XI, Subsection IWL. For corrective action, the types of corrective measures that will be considered (e.g., retensioning, tendon replacement, or reanalysis) need to be described, pending receipt and staff review of this additional information. This was Open Items 4.5.2-1.

After a number of discussions with the applicant, in a letter to the NRC dated March 14, 2001, the applicant provided the following additional information:

[*Prestress Forces Monitoring and Trending*] - The tendon surveillance is conducted every five years as required by ASME, Section XI, Subsection IWL. Trending is accomplished as required by 10 CFR 50.55a(b)(2)(ix)(B). The requirements for tendon surveillance and tendon force graphs for ANO-1 are documented and controlled in the site tendon surveillance program procedures. The IWL Inspection Program provides for the random selection of tendons. The surveillance of the selected tendons includes the following activities: inspection of the tendon components, analysis of wire and grease samples, inspection of the concrete around the tendons anchorage, and determining residual tendon force.

During the surveillance, lift-off forces for the tendons are measured and evaluated for adequacy as required by IWL. Graphs for each group of tendons (hoop, dome, and vertical tendons) provide the age related expected normalized tendon force plotted on a log-normal graph. These graphs are developed based on the tendon group and the aging effects on the reactor building concrete properties, the wire properties, and the initial prestress force. The lift-off values obtained during tendon surveillance are plotted on the graphs and trended to determine if the tendon system is performing as expected.

[*Acceptance Criteria*] - The acceptance criteria are included in the site procedures for the reactor building tendon surveillance and concrete inspections. The tendon force graphs are compared with the actual forces found during the surveillance to determine if the residual prestress in the reactor building meets the minimum required prestress.

The minimum required tendon force for each of the tendon groups is 1233 kips for the hoop tendons, 1274 kips for the vertical tendons, and 1252 kips for the dome tendons. Corrective actions will be taken should the projected tendon force for a tendon group fall below the minimum required value before the next scheduled tendon surveillance.

[*Corrective Actions*] - Conditions that do not meet the acceptance criteria in the site procedures are documented in the site condition reporting system. Evaluations are performed and acceptability is determined. Corrective actions that are needed are tracked to completion through the site condition reporting system.

Should trending indicate that prestress in a tendon group may be inadequate to meet the minimum required prestress before the next scheduled tendon surveillance, action will be taken to correct the problem. This may include re-tensioning, replacing tendons, or reanalysis of the reactor building to assure adequate prestress to meet design requirements.

The staff found this additional information acceptable to resolve Open Item 4.5.2-1.

After its initial review, the staff requested that the FSAR Supplement include a summary description of the applicant's prestress monitoring and trending activities, the acceptance criteria, and corrective actions when acceptance criteria are not met. This was FSAR Item 4.5.5 of Open Item 3.3-1.

In its revised summary description of Sections 16.2.3.6 and 16.3.4 of the FSAR Supplement, the applicant includes a description that adequately summarizes the prestress monitoring and trending activities, the acceptance criteria, and corrective actions as described above for managing prestress tendons of the ANO-1 containment in the FSAR Supplement consistent with 10 CFR 54.21(d). The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, acceptable.

4.5.3 Conclusions

On the basis of the review described above, the NRC staff has determined that there is reasonable assurance that the applicant has evaluated the TLAAs for prestress tendon force for the containment structure in accordance with 10 CFR 54.21(c)(1).

4.5.4 References for Section 4.5

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One – Unit 1, License Renewal Application," January 31, 2000.

4.6 Reactor Building Liner Plate Fatigue Analysis

Fatigue associated with the reactor building liner plate has been identified in the ANO-1 LRA as a TLAA. This section of the report documents the staff's safety evaluation of the fatigue TLAA for the reactor building liner plate, based on the information presented in Section 4.6 of the LRA.

4.6.1 Technical Information in the Application

In the LRA, Section 4.6, "Reactor Building Liner Plate Fatigue Analysis," the applicant presents the results of the fatigue TLAA for the reactor building liner plate and piping penetrations. The interior surface of the reactor building is lined with welded carbon steel plate to provide an essentially leak tight barrier. The applicant states that design criteria are applied to the liner to assure that a specified leak rate is not exceeded under DBA conditions. "Reactor Building Liner Plate and Penetrations – Fatigue," is listed in the LRA, Table 4.1-1, "List of ANO-1 Time Limited Aging Analyses."

In the LRA, Section 4.6, the applicant lists the following fatigue conditions, as described in UFSAR, Section 5.2.1.4.7.3, that were considered in the CLB design of the liner plate:

- 40 thermal cycles corresponding to 40 years of annual outdoor temperature variations
- 500 thermal cycles corresponding to reactor building interior temperature variations during reactor coolant system startup and shutdown
- one thermal cycle corresponding to DBA conditions

The design analysis of the liner plate, which considers these fatigue conditions, is considered to be a TLAA for the purposes of license renewal.

The applicant evaluates each of the above fatigue conditions for continued operation for up to 60 years. For the thermal cycles corresponding to annual outdoor temperature variations, the increase in the number of cycles from 40 to 60 is considered to be insignificant. For the thermal cycles corresponding to reactor building interior temperature variations, based on ANO-1 operating experience, the projected cycles for 60 years of operation was determined to be less than the original 500 cycle design assumption. For the thermal cycles corresponding to DBA conditions, the assumed value is considered to remain valid for 60 years of operation.

The applicant also considers additional load cycles on the liner caused by the integrated leak rate tests. Due to the limited number of these tests, the additional load cycles were stated to be bounded by the 500 cycle startup and shutdown fatigue condition.

The applicant states that the design of the reactor building piping penetrations meets the general requirements of the ASME Boiler & Pressure Vessel Code Section III for thermal cycling. Also, by design, the liner plate penetrations are isolated from thermal load cycles in the piping by concentric sleeves between the pipe and liner plate.

The applicant identifies the feedwater and main steam lines as high-temperature lines penetrating the reactor building wall and the liner plate. The applicant states that the design number of thermal load cycles in these two systems is greater than the design number of heatup and cooldown cycles of the reactor coolant system. The applicant further states that the projected number of cycles for ANO-1 through 60 years of operation has been determined to be less than the original design assumptions.

The applicant concludes that the assumed fatigue conditions used in the reactor building liner plate fatigue analysis are bounding for 60 years of plant operation. Therefore, this TLAA remains valid for the period of extended operation and meets the criteria of 10 CFR 54.21(c)(1)(i).

4.6.2 Staff Evaluation

In the LRA, Section 4.6, the applicant describes four cyclic-loads that could affect the results of the original fatigue evaluation of the containment liner plate for the period of extended operation. The applicant concludes that extrapolation of these loads from 40 to 60 years would not have a significant effect on the fatigue of the containment liner plate and that the existing fatigue analysis remains valid. The staff evaluated the information contained in LRA Section 4.6 and found it to be insufficient to support this conclusion.

The staff noted that there is no discussion of containment pressure cycling due to integrated leak rate testing. Pressure cycling and thermal load cycling may have significantly different effects on the liner plate state of stress. It is not evident from the discussion in Section 4.6 of the LRA as to how this is considered for the period of extended operation. Also, there is no definition of the projected number of these pressure cycles through the period of extended operation. To complete the review of fatigue for the liner plate, additional information on cyclic loading due to both pressure and temperature was requested in a letter to the applicant dated May 5, 2000.

In the LRA, Section 4.6, the applicant states that the number of heatup and cooldown cycles assumed in the design basis (500) envelopes the number of such cycles projected through the extended period of operation. In a letter to the applicant dated May 5, 2000, the staff requested justification for this statement.

In the LRA, Section 4.6, the applicant does not provide any information on the actual pressure and temperature cycles which are included in the calculation of cumulative fatigue usage factors for any of the penetrations through the liner plate. To complete its review, the staff requested a definition of the events, the number of occurrences assumed for design, and the projected number of occurrences through the period of extended operation for each penetration subjected to cycling loading. This information was requested of the applicant in the May 5, 2000, letter.

On the basis of the information provided in the LRA, Section 4.6, the main steam and feedwater line penetrations appear to be subject to the greatest number of thermal load cycles. There is no discussion of the effects of pressure cycling in these lines, which may also induce cyclic stresses in the penetrations. The UFSAR figure which shows the details of these penetrations was reviewed. The evaluation boundary between the liner plate penetration and the piping was not obvious and needed to be defined. This information was requested of the applicant in the May 5, 2000, letter.

In its September 7, 2000, letter to the NRC, the applicant submits supplementary information on cycle loading due to pressure and temperature. The applicant's response essentially restated what is already contained in Section 4.6 of the LRA. During a telephone conference with the applicant on October 13, 2000, the staff requested additional clarification on how past operating experience justifies the conservatism of the design-basis heatup-cooldown cycles (500) for the period of extended operation, and additional justification of their statement that pressure cycling due to integrated leak rate testing is not applicable to cumulative fatigue. In a letter to the NRC dated November 2, 2000, the applicant addresses these questions. The applicant states that within the last ten years, ANO-1 has experienced an average of approximately one heatup and cooldown per year. Thus, assuming three heatup/cooldown cycles each year through the period of extended operation is conservative, and results in a total number of cycles well below the design-basis number. The staff found the applicant's assessment acceptable.

With regard to pressure cycling, the applicant states that fatigue due to integrated leak rate testing pressure cycling loads are implicitly accounted for in the fatigue analysis by the bounding number of thermal cycles specified for fatigue evaluation. The staff notes that although pressure cycling and thermal load cycling would produce different states of stress in the liner plate, the number of cycles associated with leak rate testing is small and would have a minimal contribution to the fatigue usage factor through the period of extended operation. Since the analysis was based on a conservative number of thermal cycles, the staff's concerns were resolved.

In the same response, the applicant states that the applicable TLAAs are limited to the main steam lines and main feedwater lines mechanical penetrations. The loading conditions for these penetrations are the same as those defined in the ANO-1 UFSAR for the liner plate, namely, 500 thermal cycles of RCS startup and shutdown. This number of cycles bounds the projected number of heatup and cooldown cycles for the RCS, and is therefore acceptable. The staff found this response acceptable.

The applicant's response concerning the evaluation boundary for the main steam and feedwater line penetrations states that the evaluation boundary for mechanical penetrations, which includes the main steam and feedwater line penetrations, consists of the penetration assembly, and the weld to the process piping, but does not include the process piping within the penetrations. The staff found this response acceptable.

In a subsequent telephone conference with the applicant on October 19, 2000, concerning details of the penetration fatigue analyses, the applicant described the original fatigue analysis, which considered both through wall thermal gradients in the penetration nozzles and piping expansion loads, induced by heatup and cooldown cycling. This analysis was similar to other penetration fatigue analyses previously found acceptable by the staff. The number of heatup and cooldown cycles was shown to envelop the number of cycles projected through the extended period of operation previously discussed. Therefore, the original fatigue evaluation is considered valid for the period of extended operation. On the basis of this information, the staff considers this concern resolved.

4.6.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that, pursuant of 10 CFR 54.21(c)(1)(i), the existing fatigue TLAA for the containment liner plate and piping penetrations remains valid for the period of extended operation.

4.6.4 References for Section 4.6

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
1. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
2. "Arkansas Nuclear One - Unit 1, License Renewal Application," January 31, 2000.

4.7 Aging of Boraflex in Spent Fuel Pool Racks

Aging of Boraflex in the spent fuel pool racks plate has been identified in the LRA as a TLAA. This section of the report documents the staff's safety evaluation of the TLAA for aging of Boraflex, based on the information presented in Section 4.7 of the LRA.

4.7.1 Technical Information in the Application

In the LRA, Section 4.7, the applicant describes the TLAA for the degradation of Boraflex, which is currently used in the ANO-1 Region I spent fuel storage racks as a neutron absorber. The applicant states that the potential stressors for the Boraflex in the pool include the chemical environment of borated water and gamma radiation, which changes the material characteristics of the base polymer.

The applicant references the following NRC Information Notices (IN) and Generic Letter (GL) that identified the concern of aging of Boraflex neutron-absorbing material:

- IN 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks"
- IN 93-70, "Degradation of Boraflex Neutron Absorber Coupons"
- IN 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks"
- GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks"

In the response to Generic Letter 96-04, the applicant commits to continue monitoring and performing analyses of the Boraflex degradation at ANO-1. In the LRA, Section 4.7, the applicant states that it will continue the existing coupon monitoring program as required into the period of extended operation. The applicant also commits to the continued monitoring of spent fuel pool silica levels, and performing silica evaluations. These evaluations are based on the EPRI RACKLIFE system or its equivalent. Projected Boraflex performance will be assessed to confirm that a 5-percent subcriticality margin will be maintained as required.

The applicant states that degradation of Boraflex is treated as a TLAA at ANO-1 because it meets the six criteria of 10 CFR 54.3. In addition, the analysis meets 10 CFR 54.21(c)(1)(ii) and the sampling actions meet 10 CFR 54.21(c)(1)(iii). On the basis of these activities, the applicant concludes that the TLAA is valid for the period of extended operation.

4.7.2 Staff Evaluation

In order to determine whether the TLAA meets the requirements of 10 CFR 54.21(c), the staff reviewed the applicant's response to GL 96-04.

In the response to GL 96-04, the applicant states that long-term and accelerated test location coupon specimens are periodically removed and inspected. It is stated that "the inspections provide an indication of the general condition of the Boraflex, including gross or unusual degradation." Long-term coupons are tested approximately every five years, while accelerated coupons are tested after each refueling. The applicant also stated that it will continue to monitor spent fuel pool silica levels, perform silica evaluations based on the EPRI RACKLIFE system or its equivalent, and assess projected Boraflex performance to confirm that the 5-percent subcriticality margin will be maintained through the next evaluation period. The applicant has committed to continuing these assessments each refueling cycle prior to fuel receipt.

On the basis of its review of the information in Section 4.7 of the LRA, and the applicant's response to GL 96-04, the NRC staff cannot conclude that the effects of aging will be adequately managed for the period of extended operation. It was unclear as to the frequency of inspection and testing will be during the period of extended operation and whether there will be sufficient long-term and accelerated coupons to continue the existing monitoring program. In addition, it was unclear as to the physical condition of the coupons that were observed during inspection, and whether gap formation, and a decrease in boron density will be monitored. The applicant also did not describe the current trending analyses that have been obtained by use of the RACKLIFE code, and whether these results demonstrate that a 5-percent subcriticality margin of the spent fuel racks will be maintained for the period of extended operation. If not, the applicant needs to describe the corrective actions that will be implemented to ensure that the 5-percent subcriticality margin will be maintained through the period of extended operation.

In order to complete the evaluation of this TLAA, the staff requested additional information in a letter to the applicant dated May 5, 2000. In a letter to the NRC dated September 7, 2000, the applicant states that since the submittal of the ANO-1 LRA, Boraflex monitoring has revealed that the Boraflex is degrading more rapidly than expected. This condition has been documented in accordance with the onsite Appendix B corrective action program, and is currently being evaluated in order to determine the appropriate action. It has been determined that the Boraflex, as incorporated in the initial spent fuel pool rack design, will not last through the current 40-year licensing term, and therefore, should no longer be considered a TLAA with respect to license renewal. The applicant is evaluating several options including a revised criticality analysis, a modification of the existing spent fuel pool racks with a different neutron absorber, or a combination thereof. The applicant plans to complete the evaluation, and identify a corrective action plan for the remainder of a 60-year operating term by the fourth quarter of 2002, and plans to submit a license amendment in accordance with 10 CFR 50.90. The applicant is scheduled to complete the ANO-1 license renewal process by January 2002. Therefore, the final resolution of this concern for the entire term of the operating license at the time of submittal will be subject to NRC review and approval in accordance with 10 CFR 50.90.

The staff disagreed with the applicant's conclusion. Irrespective of the results of the condition monitoring, the Boraflex design appears to meet the definition of a TLAA in 10 CFR 54.3, as was originally stated in the application. While the applicant may continue to pursue various corrective actions in the future, the applicant will continue to rely on monitoring, evaluation, and design criteria to decide on the extent and timing of corrective actions so that the spent fuel pool design will maintain the structural and criticality design margins in accordance with the CLB. Therefore, the staff concluded that the applicant needed to provide the basis upon which the staff can conclude that there is reasonable assurance that the effects of aging of Boraflex will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(c)(1). This was Open Item 4.7.2-1.

In a letter to the NRC dated March 14, 2001, the applicant acknowledges the analysis of Boraflex in the spent fuel storage racks as a time limited aging analysis. The applicant further states that the existing analysis is not valid through the license renewal period and cannot be acceptably projected to the end of the license renewal period (as discussed in its letter to the NRC dated September 6, 2000). The applicant also agrees to continue its boraflex monitoring program to provide reasonable assurance that the effects of aging on the intended function will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). As a result of crediting the boraflex monitoring program in response to this concern, the applicant also provides the following information regarding the boraflex monitoring program requested by the staff in a letter to the applicant dated May 5, 2000:

- The applicant states that the frequency of the inspection and testing will be the same for the extended period of operation as stated in its response to Generic Letter 96-04 (0CAN109605), where Entergy Operations committed to continued monitoring and analysis of Boraflex degradation at ANO-1. The applicant states that it will continue the coupon monitoring program into the period of extended operation. Although all the accelerated coupons have been used and are no longer available, the long-term coupons are tested once every 5 years. These coupons have provided indications of Boraflex degradation. Entergy Operations will continue to monitor spent fuel pool silica levels and perform silica evaluations once per cycle. These evaluations are based on the EPRI RACKLIFE system. Boraflex performance will be projected to confirm the 5 percent subcriticality margin will be maintained as required.

- There are a sufficient number of long-term coupons to continue the existing program through the period of extended operation. The portion of the program for which the accelerated coupons were designed is complete.

- ANO-1 currently has a procedure in place for examining and testing the spent fuel pool Boraflex test coupons. The coupon inspections consist of taking thickness measurements, density determinations, general visual inspection, and hardness testing. Neutron attenuation testing is also performed on the sampled Boraflex coupons. This testing more accurately determines areal Boron density.

- The minimum as-designed Boraflex dimensions and the minimum designed areal Boron 10 densities with an assumed degradation of 10 percent are used in the ANO-1 criticality analysis. The ANO-1 criticality analysis assumes all the shrinkage is on the ends, which is more conservative than gap formation assumptions for the ANO-1 rack geometry. The Boraflex panels are assumed to shrink 4.1 percent in width. Current RACKLIFE analysis indicates that there is less than 10 percent boron degradation.

  These assumptions and analytical calculations have been correlated to industry data obtained through in-situ testing of a similar rack design to ANO-1. The results from the tested racks are conservatively applied to the ANO-1 racks based upon the tested racks having been subjected to higher doses, the spent fuel pool silica levels exceeding the concentrations seen at ANO-1, and the tested racks have a higher peak panel degradation. The tested racks have also been in service longer than the ANO-1 racks.

- The results of the current Boraflex trending analysis demonstrate that the 5 percent subcriticality margin is being maintained; however, it will not be maintained for the period of extended operation. As previously discussed, this condition has been documented in accordance with the onsite Appendix B corrective action program. Corrective actions will be implemented to ensure that the 5 percent subcriticality margin will be maintained through the period of extended operation. Corrective actions may include modification of the spent fuel racks to incorporate a different neutron absorber material. Entergy Operations is committed to resolving this issue as documented in correspondence dated September 6, 2000 (1CAN090002).

The staff found this resolution to Open Item 4.7.2-1 acceptable.

After its initial review, the staff also requested that the FSAR Supplement include a summary description of the applicant's monitoring, evaluation activities, optional corrective actions, and decision criteria for the aging of Boraflex in the spent fuel pool. This was FSAR Item 4.7.3 of Open Item 3.3-1.

In its revised summary description of Section 16.3.6 of the FSAR Supplement, the applicant included a description of the monitoring, evaluation activities, optional corrective actions, and decision criteria for the aging of Boraflex in the spent fuel pool consistent with the information described above. The staff finds the revised summary description as submitted by the applicant in a letter to the NRC dated March 14, 2001, to be acceptable and, therefore, finds FSAR Item 3.3.1.2.3 of Open Item 3.3-1 resolved.

4.7.3 Conclusions

On the basis of the review described above, the staff finds that the applicant has demonstrated that there is reasonable assurance, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the intended function of Boraflex will be adequately managed for the period of extended operation.

4.7.4 References for Section 4.7

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. DG-1047, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, April 21, 2000.
3. "Arkansas Nuclear One – Unit 1, License Renewal Application," January 31, 2000.
4. NRC IN 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks."
5. NRC IN 93-70, "Degradation of Boraflex Neutron Absorber Coupons."
6. NRC IN 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks."
7. NRC GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks."

4.8 Other Time-Limited Aging Analyses

The TLAAs evaluated in this section of the SER include the following:

- reactor vessel underclad cracking
- reactor vessel incore instrumentation nozzle FIVs
- leak-before-break in RCS piping
- reactor coolant pump motor flywheels

4.8.1 Reactor Vessel Underclad Cracking

In the LRA, Section 4.8, the applicant discusses the TLAA for the intergranular separations.

4.8.1.1 Technical Information in the Application

In this TLAA, the applicant addresses the issue of intergranular separations (underclad cracking) in low-alloy steel heat-affected zones under austenitic stainless steel weld cladding in SA 508, Class 2 reactor vessel forgings with coarse grain structures. The applicant references the topical report BAW-10013, "Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Cladding," as containing a fracture mechanics calculation that demonstrates that the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size plus predicted flaw growth due to fatigue cycling. The analysis concluded that intergranular separation in B&W vessels would not lead to failure. This conclusion, according to the applicant, was accepted by the Atomic Energy Commission. To cover the period of extended operation, the applicant performs a calculation using current ASME Code requirements. This analysis is given in Appendix C to BAW-2251A.

4.8.1.2 Staff Evaluation

The NRC staff has evaluated the B&WOG approach to resolving underclad cracking issues in Appendix C to the final SER for BAW-2251A. The B&WOG approach includes the following conservatisms:

- a maximum crack depth of 0.165 inch reported by industry as the initial crack depth, instead of the 0.1 inch size reported for reactor pressure vessels

- a total effective crack depth of 0.353 inch (nominal cladding thickness of 0.1875 inch plus 0.165 inch crack depth in the underlying alloy steel)

- assumption that all the cracks are surface cracks

- use of the fatigue crack growth rate for surface flaws in a water reactor environment

- use of a safety factor of 17 percent more than that specified by the ASME Code for Levels A and B loading, and 72 percent more for Levels C and D loading

The maximum crack growth and applied stress intensity factor for the normal and upset conditions were found to occur near the nozzle belt region. The maximum crack growth considering all of the normal and upset transients for 48 EFPY was determined by B&WOG to be 0.180 inch. This gives a final crack depth of 0.533 inches (0.353 inch plus 0.180 inch). The maximum applied stress intensity factor for the normal and upset conditions results in a fracture toughness margin of 3.6, which is greater than the ASME IWB-3612 acceptance criterion of 3.16. The maximum applied stress intensity factor for the emergency and faulted conditions was shown by B&WOG to be in the closure head to head flange region and the fracture toughness margin was found to be 2.24, which is greater than the ASME IWB-3612 acceptance criterion of 1.41.

The NRC staff found that, consistent with the final SER for BAW-2251A, the B&WOG underclad cracking flaw analysis, performed in accordance with 10 CFR 54.21(c), is acceptable for the period of extended operation.

4.8.1.3 Conclusions

On the basis of its review, the staff concludes that the B&WOG's underclad cracking flaw analysis satisfies the requirements of 10 CFR 54.21 (c).

4.8.2 Reactor Vessel Incore Instrumentation Nozzle – FIV Endurance Limit

In the LRA, Section 4.8.2, the applicant presents a description of a TLAA for flow-induced fatigue in the reactor vessel incore instrumentation nozzles.

4.8.2.1 Technical Information in the Application

The applicant discusses the evaluation of FIV of the reactor vessel incore instrumentation nozzles. For the current licensing period, BAW-10051, "Flow Induced Vibration Endurance Limit Assumptions," contains an analysis of the stresses in the reactor vessel incore instrumentation nozzles. The topical report compares the stresses to the fatigue endurance limits. However, the analysis does not cover the period of extended operation.

4.8.2.2 Staff Evaluation

In the initial calculation, given in BAW-10051, the endurance limits were based on 10^{12} cycles over 40 years. The applicant states that the fatigue cycles were extended to 60 years in a new calculation. The component stress values were acceptable, when compared to the newly calculated endurance limits for 60 years.

In a letter to the NRC dated September 6, 2000, the applicant describes the new analysis that covers the period of extended operation. In the analysis, the number of fatigue cycles over the 60-year operating period was conservatively assumed to be 10^{13}, which is an order of magnitude greater than that estimated for the current 40 year licensing period. The fatigue endurance limit was assumed to be reduced by 4 percent for each decade of cycles. This assumption is consistent with that given in BAW-10051, Appendix A. Thus, the applicable fatigue endurance limit from ASME III, Division 1, which extends only to 10^{11} cycles, is reduced by a factor of $(0.96)^2$ at 10^{13} cycles. A correction factor of 0.9 is also applied to the ASME fatigue curve. The fatigue curve is for room temperature, and the correction factor is added to account for the reduction in Young's modulus at the nozzle operating temperature. The applicable endurance limit at 10^{13} cycles becomes 13,700 psi. A similar calculation was carried out for high strength bolting and gave an endurance limit of 9,100 psi. From Table 5.1 of BAW-10051, the alternating stresses for the incore instrumentation nozzles and bolting were shown to be at least 19 percent lower than the calculated endurance limits at 10^{13} cycles. This indicates that fatigue failure is highly unlikely. On the basis of the methodology used and the conservative results from implementing this methodology, the staff found the applicant's TLAA evaluation of the reactor vessel incore instrumentation nozzles acceptable for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.8.2.3 Conclusions

On the basis of its review, the staff concludes that the applicant's FIV analysis satisfies the requirements of 10 CFR 54.21(c)(1)(ii).

4.8.3 Leak-Before-Break

The applicant's leak-before-break (LBB) analysis is given in Section 4.8.3 of the application.

4.8.3.1 Technical Information in the Application

The application describes the LBB approach for the RCS main coolant loop piping.
It is based on the analysis given in topical report BAW-1847, Revision 1, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS." This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted conditions and was approved by the staff for the current licensing period.

The LBB analyses described in the LRA include the following items:

- fatigue flaw growth
- thermal aging effects in cast austenitic stainless steel (CASS) reactor coolant pump (RCP) inlet and discharge nozzles

For fatigue flaw growth, the applicant uses an analysis to show that the number of fatigue cycles that were originally defined for the current period of operation will not be exceeded during the period of extended operation. In the case of thermally-induced embrittlement of CASS RCP inlet and discharge nozzles, the applicant calculates the maximum anticipated crack lengths in the nozzles, and estimated the margin of safety. The two analyses are reviewed in the next section of this SER.

4.8.3.2 Staff Evaluation

Fatigue Flaw Growth

The LBB analysis, described in BAW-1847, Revision 1, was carried out in accordance with guidance given in NUREG 1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks." Specifically, a surface flaw is postulated at selected locations of the piping system (i.e., highest stress coincident with the lower bound of materials properties for base metal, welds, and safe-ends). The analysis seeks to demonstrate that such a surface flaw will propagate through the wall and cause an identifiable leak, before it can propagate circumferentially around the pipe to such an extent that it could cause a double-ended pipe rupture under faulted conditions.

In the LRA, Section 4.3.5, the applicant describes its program to monitor the number of transients for the current licensing period. The applicant intends to undertake corrective actions if the number of transient cycles exceeds the allowable design limit. Currently, the design limit, as given in Section 4.3, Table 4-3, of BAW-1847, Revision 1, is 240 heatup and cooldown cycles and 22 cycles of safe shutdown earthquake. The applicant states that the flaw growth evaluation in BAW-1847, Revision 1, is applicable to 60 years of operation since it has not revised the transients defined in the RCS design specifications for the period of extended operation.

The staff found the TLAA acceptable since it covers the period of extended operation, in accordance with the requirements of 10 CFR 54(c)(1)(i).

Thermal Aging of CASS Reactor Coolant Pump Suction and Discharge Nozzles

The applicant identifies thermal aging of CASS components as a potential problem with respect to maintenance of sufficient piping material fracture toughness. The applicant references the review in BAW-1847, Revision 1, and NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels." The latter report showed that prolonged heating of CASS to reactor coolant temperatures could lead to a loss in fracture toughness.

In a letter to the applicant dated May 5, 2000, the staff requested assurance that the δ-ferrite content of the CASS RCP nozzles were within the bounds of applicability of data in NUREG/CR-6177, which gives guidance on LBB analyses for CASS components. In its response to the NRC dated September 6, 2000, the applicant states that the δ-ferrite content of the CASS RCP nozzles was 14.2 percent, and that this was within the bounds reported in NUREG/CR-6177. A review of NUREG/CR-6177 confirmed that the normal ferrite content of domestic CASS is ≤15 percent.

The applicant provides a flaw stability analysis in Section 4.8.3 of the LRA to show the acceptability of the LBB concept for the RCS main coolant piping over the period of extended operation. The analysis was performed on suction and discharge nozzles of the RCP casings since the applicant states that they were susceptible to loss of fracture toughness due to thermal aging. In the analysis, the lower bound CASS fracture toughness properties were used.

In the applicant's analysis, bounding 10 gpm crack sizes (margin of 10 on the plant's leak detection capability) for the RCP suction and discharge nozzles were determined using a method consistent with that reported in BAW-1847, Revision 1. In the revised analysis, the applied loadings were considered, using the absolute sum load combination method. The leakage crack length (twice the leakage flaw size) for the suction nozzle was determined to be 8.62 inches, and for the discharge nozzle it was found to be 8.86 inches. In addition, a crack extension value of 0.6 inches was considered in the flaw stability analysis. The flaw stability analysis was performed for the suction and discharge nozzles for the reactor coolant pump. The discharge nozzle was found to be bounding. The critical crack length was found to be 21.6 inches. Therefore, the margin was determined to be 2.4.

The applicant describes an additional calculation in the LRA for crack propagation in thermally-aged SMAW material connecting the stainless steel transition pieces to the RCP nozzles since the structure of the welds is similar to CASS. Using data from the literature, it was shown that the LBB analyses for aged CASS material bounds that for the SMAW material.

On the basis of the large margin of safety on the calculated critical crack length for CASS RCS components, the staff found the above analysis, conducted in accordance with 10 CFR 54.21(c)(1)(i), to be acceptable.

4.8.3.3 Conclusions

The staff accepts the TLAA regarding LBB, fatigue flaw growth, and thermal aging of the CASS RCP inlet and discharge nozzles as demonstration that the appropriate bases for LBB will be maintained through the extended period of operation for ANO-1. On the basis of its review, the staff finds that the applicant's LBB analysis satisfies the requirements of 10 CFR 54.21(c) (1)(i).

4.8.4 Reactor Coolant Pump Motor Flywheels

The applicant evaluates the TLAA relating to fatigue of the reactor coolant pump (RCP) motor flywheel in Section 4.8.4 of the LRA.

4.8.4.1 Technical Information in the Application

The RCP motors are large, vertical, squirrel cage motors. The motors have flywheels to increase rotational inertia thus prolonging pump coastdown, and ensuring a more gradual loss of main coolant flow to the core in the event pump power is lost. The aging mechanism of concern is fatigue crack growth of pre-existing cracks in the flywheel bore keyway from stresses due to starting the motor. Therefore, this topic is considered a TLAA for license renewal. The applicant addresses the TLAA by projecting the existing analysis to the end of the period of extended operation.

4.8.4.2 Staff Evaluation

The applicant references a crack growth evaluation, which shows that crack sizes remain acceptable for 4,000 startup/shutdown cycles. This number of cycles is reported in the LRA to exceed the number of design cycles by a factor of 8. The applicant identifies that the RCP pump starts normally occur once every 200 to 300 days, on average; this conservative design is considered valid for the period of extended operation. On this basis, the NRC staff concludes that the applicant has provided an acceptable basis for extending the TLAA for the RCP flywheel to cover the period of extended operation and, therefore, meets the requirements of 10 CFR 54.21(c)(1)(ii).

4.8.4.3 Conclusions

The staff concludes that the applicant has provided an acceptable TLAA involving components of the RCP flywheel as defined in 10 CFR 54.3 and meets 10 CFR 54.21(c)(1)(ii).

4.8.5 References for Section 4.8

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, Draft Regulatory Guide DG-1047.
3. "Arkansas Nuclear One – Unit 1, License Renewal Application," January 31, 2000.
4. BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibrations,"September, 1972.
5. BAW-10013, "Study of Intergranular Separations in Low-Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding," B&W Nuclear Power Generation, December 1971.
6. "ASME Boiler and Pressure Vessel Code," American Society of Mechanical Engineers.
7. BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," The B&W Owners Group Generic License Renewal Program, June 1996.
8. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers.
9. BAW-1847, "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS," Revision 1, B&W Owners Group, September 1985.
10. BAW-2243A, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping," The B&W Owners Group Generic License Renewal Program, June 1996.

5 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During the 480th meeting of the Advisory Committee on Reactor Safeguards (ACRS) on March 1, 2001, the ACRS reviewed the NRC staff's safety evaluation report (SER) related to the license renewal application (LRA) for Arkansas Nuclear One, Unit 1 (ANO-1). The ACRS Subcommittee on Plant License Renewal initially reviewed the SER prior to its meeting with the NRC staff and the applicant on February 22, 2001, and presented its findings during the March 1, 2001 ACRS meeting. Because of the small number and subject matter of the open items, the subcommittee recommended not issuing an ACRS interim letter on its review of the ANO-1 license renewal SER with open items. The subcommittee also recommended that the final SER be presented directly to the ACRS without a separate, second subcommittee meeting to review the resolution of the six open items. The staff submitted the final SER related to the LRA for ANO-1 with the resolution to the open items on April 12, 2001. The staff briefed the ACRS full-committee on May 10, 2001, regarding the resolution of open items.

During the 482th meeting of the ACRS on May 10, 2001, the ACRS completed its review of the ANO-1 LRA, and documented its findings in a letter dated May 18, 2001. A copy of that letter is provided within.

May 18, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR ARKANSAS NUCLEAR ONE, UNIT 1

During the 482nd meeting of the Advisory Committee on Reactor Safeguards, May 10-11, 2001, we completed our review of Entergy Operations, Inc., application for license renewal of Arkansas Nuclear One, Unit 1 (ANO-1), and the related final Safety Evaluation Report (SER). Our review included two meetings with the staff and the applicant. We had the benefit of the documents referenced.

Conclusions and Recommendations

1. Entergy has properly identified the structures, systems, and components (SSCs) that are subject to aging management review consistent with the requirements of 10 CFR Part 54.

2. Aging mechanisms associated with passive, long-lived SSCs have been appropriately identified.

3. The programs instituted to manage aging-related degradation of the identified SSCs are appropriate and provide reasonable assurance that ANO-1 can be operated in accordance with its current licensing basis for the extended license term without undue risk to the health and safety of the public. The programs do not explicitly address the potential for circumferential cracking in control rod drive mechanism (CRDM) nozzle penetrations, such as has been observed at the Oconee Nuclear Plant, Unit 3. We expect that this current problem will be resolved and that the resolution will be incorporated into the current licensing basis and carried over into the license renewal period.

4. The staff has performed a comprehensive and thorough review of Entergy's application, and the open items identified in the January 2001 draft SER have been satisfactorily resolved .

5. The staff should determine whether modification of the current guidance in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," is required to reflect the lessons learned from the ANO-1 application regarding aging management of small-bore piping and medium-voltage buried cable.

Background and Discussion

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. Entergy requested renewal of the operating license for ANO-1 for a period of 20 years beyond the current license term, which expires on May 20, 2014. The final SER documents the results of the staff's review of information submitted by Entergy, including those commitments that were necessary to resolve open items identified by the staff in its January 2001 draft SER. The staff's review included verification of the completeness of the SSCs identified in the application, the validation of the integrated plant assessment process, the identification of the possible aging mechanisms associated with each passive long-lived component, and the adequacy of the aging management programs.

Our Subcommittee on Plant License Renewal met with the applicant and the staff on February 22, 2001, to review the SER with open items. The Subcommittee did not identify any issues to be addressed other than the six open items identified by the staff. This remarkably small number of open items is due, in large part, to the fact that the applicant implemented relevant lessons learned from the previous license renewal applications. In addition, the applicant structured the application using the standard application format and the guidance in Nuclear Energy Institute (NEI) Report 95-10, which facilitated the review. Because of the small number of open items and the scrutability of the application, we decided that there was no necessity to provide an interim report and have reviewed the SER on an accelerated basis.

The process implemented by the applicant to identify SSCs within the scope of the License Renewal Rule is effective. Reactor coolant system (RCS) components were identified using the generic Babcock & Wilcox Owners Group (BWOG) topical reports that address aging of RCS piping, pressurizer, reactor vessel, and reactor vessel internals. These topical reports, which have been approved by the staff, are applicable to ANO-1 and were used to support the license renewal application for Oconee. All other components in scope were determined on a plant-specific basis. At ANO-1, the safety-related SSCs included in the quality assurance program ("Q" list), as required by 10 CFR Part 50, Appendix B, are those that meet the definition of "safety related" in 10 CFR 54.4(a)(1). Furthermore, the majority of SSCs whose failure could prevent satisfactory accomplishment of any of the safety-related functions in 10 CFR 54.4(a)(1) are also classified as safety-related and included in the ANO-1 "Q" list. Therefore, the applicant was able to use the "Q" list to identify the bulk of the ANO-1 SSCs within the scope of the License Renewal Rule. This process has also resulted in the conservative inclusion of some SSCs that do not meet the criteria of 10 CFR 54.4(a)(2). We concur with the staff that the applicant has properly identified SSCs requiring an aging management review.

The applicant conducted a comprehensive aging management review of SSCs in scope. Aging effects of RCS components were identified using the aforementioned BWOG topical reports. Aging effects of all other SSCs were identified based on component material, operating environment, and operating stresses using plant-specific and industry-wide operating experience. Appendix B of the application describes the 22 existing or modified programs and the seven new programs implemented to manage aging during the period of extended operation.

ANO-1 has proposed a significantly smaller number of one-time inspections than did previous applicants. This is due, in part, to the fact that existing or modified ANO-1 programs manage aging effects that previous applicants do not manage during their current license terms. Consequently, previous applicants had to implement a larger number of one-time inspections to support license renewal. For example, aging of small-bore piping is managed at ANO-1 by a plant-specific risk-informed inspection program, and therefore, does not require a one-time inspection. We agree with the staff that the applicant has properly identified possible aging mechanisms associated with passive, long-lived SSCs and that the programs instituted to manage aging degradation of the identified SSCs are appropriate.

The ANO-1 application identifies cracking at welded joints of the CRDM pressure boundary as an aging effect to be managed. Appendix B of the application describes the aging management program instituted to deal with this aging degradation mechanism; i.e., "CRDM nozzle and other vessel closure penetration inspection program." This program identifies primary water stress corrosion cracking of Alloy-600 nozzles with partial penetration welds as the aging effect of concern and ties programmatic elements, such as the frequency of inspections, to the results of plant-specific and sister plant inspection findings. The initiatives included in this program are adequate to deal with this identified aging effect during the remaining portion of the current license term and during the period of extended operation. However, it is likely that the recent observations of stress corrosion cracking at the outer surface of CRDM nozzle penetrations may require some revisions to the program. We have noted previously that aging management programs may have to be revised if it is found that new modes of degradation are occurring.

The ANO-1 application includes time limited aging analyses (TLAA) to evaluate the impact of neutron embrittlement on reactor vessel integrity. These analyses determine reactor vessel resistance to failure during pressurized thermal shock (PTS) events and the maintenance of acceptable Charpy upper-shelf energy levels. The TLAA used the methodology described in topical report BAW-2251A, "Demonstration of the

Management of Aging Effects for the Reactor Vessel." This topical report was reviewed and approved by the staff and reviewed by the ACRS. Based on the composition of the limiting welds, Entergy projected that the ANO-1 reactor vessel will not reach the PTS and Charpy upper-shelf energy screening limits until well after 60 years of operation. The ANO-1 reactor vessel integrity program will be utilized to ensure that the time-dependent parameters used in the TLAA evaluations are tracked so that the TLAA remain valid during the license renewal period.

Entergy committed to implementing a plant-specific program to manage the effects of fatigue. Using the correlations published in NUREG/CR-5704, Entergy has found that the surge line, the high pressure injection/makeup nozzles, and safe ends may reach the limits of acceptable fatigue during the period of extended operation. To address this condition, Entergy has proposed a program that will include one or more of the following options: refinement of the fatigue analyses, repair, replacement, or management of fatigue effects using a program that will be reviewed and approved by the staff. We concur with the staff that Entergy's proposed program is an acceptable plant-specific approach for resolving the concerns of Generic Safety Issue-190, "Fatigue Evaluation of Metal Components for 60 Year Plant Life."

ANO-1 region 1 spent fuel storage racks currently use Boraflex as a neutron absorber. Aging of Boraflex was identified in the application as a time limited aging analysis. During the staff's review of the ANO-1 application, Entergy informed the staff that Boraflex had been found to degrade more rapidly than previously expected, and was not expected to last through the current 40-year licensing term. Therefore, a corrective action plan for the remainder of the 60-year operating term would be identified and committed to before the end of 2002. In Open Item 4.7.2-1 associated with Boraflex degradation, the staff requested that Entergy continue to recognize aging of Boraflex as a time limited aging analysis and provide details on the required monitoring program. Entergy has now provided the requested programmatic details. We concur with the staff that either the implementation of a permanent solution during the current licensing period or the Boraflex monitoring program provided by Entergy and described in the SER provides acceptable management of Boraflex degradation during the period of extended operation.

The staff has performed a comprehensive and thorough review of Entergy's application. The applicant and the staff have identified possible aging mechanisms associated with passive long-lived components. Adequate programs have been established to manage the effects of aging so that ANO-1 can be operated safely in accordance with its current licensing basis for the extended license term.

The review of the ANO-1 application has provided significant new information on small-bore piping and medium-voltage buried cable aging degradation and related management programs. As described above, ANO-1 has implemented a small-bore piping inspection program because it has identified small-bore piping in safety-significant locations that is susceptible to aging degradation. The staff should determine whether current guidance in the GALL report needs to be modified to reflect this experience. Also, ANO-1 has implemented a medium-voltage buried cable aging management program that includes the options of cable testing or periodic replacement of buried cables. ANO-1 has included the replacement option because it has found that in a number of instances testing was not effective in identifying cable degradation. The staff needs to evaluate the adequacy of testing of buried cables and provide appropriate guidance in the next update of the GALL report.

Dr. William J. Shack did not participate in the Committee's deliberations on aging-induced degradation.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1," dated April 2001.
2. Letter dated January 31, 2000, from C. R. Hutchinson to the U.S. Nuclear Regulatory Commission, Subject: Arkansas Nuclear One, Unit 1, License Renewal Application.
3. Letter dated March 14, 2001, from J. D. Vandergrift to the U.S. Nuclear Regulatory Commission, Subject: Arkansas Nuclear One, Unit 1, License Renewal Safety Evaluation Report Open Item Responses.
4. Babcock and Wilcox Owners Group Generic License Renewal Program Topical Report, BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," dated June 1996.
5. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Steels," dated April 1999.
6. U. S. Nuclear Regulatory Commission, Generic Safety Issue - 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."

6 CONCLUSIONS

In accordance with Federal regulations under Title 10 of the *Code of Federal Regulations*, Part 51 and Part 54, and the NRC draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," dated September 1997, the staff has completed its review of the Arkansas Nuclear One, Unit 1(ANO-1) license renewal application and supporting documentation, and has documented its finding in this safety evaluation report (SER). The standards for issuance of a renewed license are set forth in 10 CFR 54.29.

In the SER issued on January 10, 2001, regarding the review of the ANO-1 license renewal application, the staff identified six open items. Those open items have been resolved, as discussed in this SER. On the basis of its evaluation of the ANO-1 license renewal application and the applicant's response to the open items as discussed within this SER, the staff concludes the following:

1. actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require an aging management review under 10 CFR 54.21(a)(1)

2. actions have been identified and have been or will be taken with respect to time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c)

Accordingly, the staff finds that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis for ANO-1. The staff notes that the results of the staff's environmental review are documented in the final plant-specific supplement to the Generic Environmental Impact Statement.

APPENDIX A
CHRONOLOGY

This appendix contains a chronological listing of routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and Entergy Operations, Inc., and other correspondence regarding the NRC staff's review of the Arkansas Nuclear One, Unit 1 (under Docket Nos. 50-313) application for license renewal.

.January 31, 2000	In a letter (signed by J. Vandergrift) Entergy submitted its License Renewal Application (LRA) for Arkansas Nuclear One, Unit 1 (ANO-1) as well as a copy of the boundary drawings to the NRC.
February 4, 2000	In a letter (signed by C. Grimes) NRC informed Entergy that the NRC received ANO-1 LRA on February 1, 2000, and that Mr. Robert J. Prato was appointed as the project manager for ANO-1 License Renewal Application.
February 14, 2000	In a letter (signed by J. Vandergrift) Entergy informed NRC that as of February 12, 2000, Mr. Craig G. Anderson replaced Mr. Randy Hutchinson as Vice President, Operations, at ANO-1.
February 28, 2000	In a letter (signed by D. Mathews) NRC informed Entergy that the NRC staff has determined that Entergy has submitted sufficient information that is complete and acceptable for docketing.
March 7, 2000	In a letter (signed by R. Prato) NRC informed Entergy of the schedule for the conduct of review of the ANO-1 LRA.
April 11, 2000	In a letter (signed by J. Vandergrift) Entergy submitted corrections to the LRA Environmental Report (ER) and also provided information on severe accident mitigation alternatives (SAMA).
April 12, 2000	In a letter (signed by R. Prato) NRC issued a public meeting notice to the stakeholders and the public and informed that a meeting to be held on May 17, 2000, with Entergy to discuss the status of review of license renewal application for ANO-1.
April 12, 2000	In a letter (signed by T. Kenyon) NRC requested Entergy for additional information (RAI) regarding severe accident mitigation alternatives for ANO-1.
April 17, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.5, 3.5, and portions of 4.4 of the ANO-1 LRA.
April 25, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 3.3.2.4, 3.3.2.5, 3.3.2.6, 3.7, 4.1, 4.2, 4.3, 4.4, 4.8.1, 4.8.2, and 4.8.3 of the ANO-1 LRA.

May 2, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.3.2.4, 2.3.2.5, 2.3.2.6, 2.3.2.7, 2.3.2.8, 2.3.3.11, 2.3.3.12, 2.3.3.13, 2.4, and 3.7 of the ANO-1 LRA.
May 5, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.2, 2.3.3.1, 2.3.3.2, 2.3.3.3, 2.3.3.4, 2.3.3.5, 2.3.3.6, 2.3.3.7, 2.3.3.8, 2.3.3.9, 2.3.3.10, 2.3.4, 3.3.2.2, 3.3.2.3, 3.3.2.7, 3.3.2.8, 4.5, 4.6, and 4.7 of the ANO-1 LRA.
June 1, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.3.1, 3.3, and 3.6 of the ANO-1 LRA.
June 5, 2000	In a letter (signed by T. Kenyon) NRC requested Entergy to provide additional information (RAI) regarding its January 2000 Environmental Report for ANO-1.
June 6, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Section 4.4 of the ANO-1 LRA requested on April 17, 2000, and April 25, 2000.
June 9, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 3.3.1.1, 3.3.1.2, and 3.3.4 of the ANO-1 LRA.
June 23, 2000	In a letter (signed by R. Prato) NRC requested Entergy to provide additional information (RAI) on Sections 2.3.1, 3.3, and 3.6 of the ANO-1 LRA.
July 6, 2000	Response to NRC staff requests for additional information regarding equipment qualification.
July 31, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 2.5 and 3.7 of the ANO-1 LRA requested on April 17, 2000, April 25, 2000, and May 2, 2000.
August 24, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 2.3.1.5, 2.3.1.6, 3.2.4, 3.2.5, and 4.2 of the ANO-1 LRA requested on April 25, 2000, and June 1, 2000.
August 30, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Section 2.0 of the ANO-1 LRA requested on May 2, 2000, May 5, 2000, June 1, 2000.

September 6, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 2.3.1.3, 2.3.1.4, 2.3.1.7, 3.2.2, 3.2.3, 3.2.6, 3.2.7, 3.2.8, 4.3, 4.7, and 4.8 of the ANO-1 LRA requested on April 25, 2000, May 5, 2000, and June 1, 2000.
September 7, 2000	In a letter (signed by J. Vandergrift) Entergy provided its response to the NRC RAIs on Sections 3.1.3, 3.6, 4.5, and 4.6 of the ANO-1 LRA requested on May 5, 2000, June 1, 2000, and June 23, 2000.
September 12, 2000	In a letter (signed by J. Vandergrift) Entergy provided its responses to the NRC RAIs on Section 3.0 of the ANO-1 LRA requested on April 17, 2000, May 5, 2000, June 1, 2000, and June 9, 2000.
October 3, 2000	In a letter (signed by J. Vandergrift) Entergy provided additional clarifications and/or corrections to its responses to the NRC RAI #s 3.3.2.2.2.2-1(d), 4.2.3-3, 2.3.1-4, 2.4-5, 3.3.3.1-7, 3.3.3.1-2(b), 3.3.3.1-6(a), and 3.3.1.4.4-2. These RAIs were requested on August 24, 2000, August 30, 2000, September 6, 2000, and September 12, 2000.
October 11, 2000	In a letter (signed by R. Prato) NRC provided the summary of conference calls between the NRC staff and members of ANO-1in order to obtain clarifying information for the Entergy's responses to the staff's RAIs. These conference calls were conducted on September 13, 2000, September 18, 2000, September 20, 2000, and October 3, 2000.
October 20, 2000	In a letter (signed by R. Prato) NRC provided the summary of conference calls between the NRC staff and members of ANO-1in order to obtain clarifying information for the Entergy's responses to the staff's RAIs. These conference calls were conducted on October 11, 2000, October 12, 2000, and October 13, 2000.
November 2, 2000	In a letter (signed by J. Vandergrift) Entergy provided additional clarifications and/or corrections to its responses to the NRC RAIs from Sections 2.3.3.2, 2.3.3.6, 3.3.2.2.2.2, 3.3.2.2.2, 3.3.2.3.2.2, 3.3.2.6.2.2, 3.3.4.3.1, 3.3.4.3.2, 3.3.4.3.2.5, 3.3.4.3.1, 3.3.4.3.1.8, 3.3.4.3.2, 3.3.4.3.2.2, 3.3.4.3.2.3, 3.3.4.3.2.9, 3.3.4.3.2.10, 2.3.3.9, 3.3.5, 3.3.6, 4.5, and 4.6.
December 4, 2000	Telecommunication for clarification of information relating to the ANO-1 LRA and Site summary visit.
December 20, 2000	Clarification to request for additional information relating to the ANO-1 LRA.

January 10, 2001	The NRC staff issued the license renewal safety evaluation report with open items for ANO-1
February 21, 2001	The NRC Staff issue notice of forthcoming public meeting with Entergy Operations, Inc., on license renewal fire protection scoping for Arkansas Nuclear One, Unit 1
February 22, 2001	ACRS Subcommittee for license renewal meet with the NRC staff and Entergy Operation, Inc., to discuss the ANO-1 license renewal applicant and the staff's safety evalution with open items
March 1, 2001	The ACRS subcommittee for license renewal met with the ACRS full committee to summarize its findings related to the ANO-1 license renewal application as was documented in the January 10, 2001 SER with open items.
March 14, 2001	In a letter (signed by J. Vandergrift) Entergy provided its responses to the license renewal Safety Evaluation Report Open Items
April 2, 2001	In a letter to the applicant, the staff sent a revised schedule reducing the duration of the ANO-1 license renewal application review from 25-months to 18-months
April 9, 2001	The NRC staff issued the final license renewal safety evaluation report for ANO-1
April dd, 2001	The final EIS regarding the license renewal of ANO-1 was issued
May dd, 2001	Regional Administrator, Region IV, submits his recommendation regarding the license renewal of ANO-1
May 10, 2001	The staff met with the ACRS full committee to summarize its findings related to the ANO-1 license renewal application as was documented in the April, dd, 2001, SER
May dd, 2001	The ACRS documents its findings regarding the ANO-1 LRA and submits its recommendation to the Commission

APPENDIX B
REFERENCE DOCUMENTS

This appendix contains a listing of references used in the preparation of the Safety Evaluation Report prepared during the review of the license renewal application for Arkansas Nuclear One Unit 1 under Docket Numbers 50-313

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Procedure NES-16, "Accident Analysis ULD and AIM Basis Document Format and Content," Revision 1.

Procedure 1000.150, "Licensing Document Maintenance," Revision2.

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USAS B31.7, "Nuclear Power Piping."

APPENDIX C
PRINCIPAL CONTRIBUTORS

NAME	RESPONSIBILITY
H. Ashar	Structural Engineering
G. Bagchi	Structural Engineering
M. Banic	Materials Engineering
W. Bateman	Technical Support
S. Coffin	Materials Engineering
J. Davis	Materials Engineering
D. Diec	Plant Systems
T. Eaton	Plant Systems (Fire Protection)
J. Fair	Mechanical Engineering
Z. Fu	Materials Engineering
G. Galletti	Quality Assurance
G. Georgiev	Structural Engineering
C. Gratton	Plant Systems
B. Grenier	Technical Support
C. Grimes	Technical Support
F. Grubelich	Mechanical Engineering
J. Guo	Plant Systems
M. Hartzman	Mechanical
C. Holden	Electrical Engineering
S. Hou	Structural Engineering
D. Jeng	Structural Engineering
M. Khanna	Materials Engineering
Y. Kim	Mechanical Engineering
C. Lauron	Structural Engineering
A.D. Lee	Mechanical Engineering
A.J. Lee	Mechanical Engineering
C. Li	Plant Systems
Y. Li	Mechanical Engineering
J. Ma	Structural Engineering
K. Manoly	Electrical Engineering
P. Milano	Safety and Environment
M. Mitchell	Materials Engineering
J. Moore	Legal Councel
C. Munson	Structural Engineering
D. Nguyen	Electrical Engineering
A. Pal	Electrical Engineering
P. Patnaik	Structural Engineering
K. Parczewski	Chemical Engineering
J. Peralta	Quality Assurance
J. Rajan	Mechanical Engineering
J. Raval	Plant Systems
K. Rico	Technical Support
J. Strosnider	Technical Support
E. Sullivan	Materials Engineering
O. Tabatabai-Yazdi	Technical Support

R. Wessman	Technical Support
K. Wichman	Technical Support

CONTRACTORS

Argonne National Laboratory	Aging Management Review
Brookhaven National Laboratory	Aging Management Review
Idaho National Engineering and Environmental Laboratory	Aging Management Review
Pacific Northwest National Laboratory	Aging Management Review

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11. ABSTRACT *(200 words or less)*

This document is a safety evaluation report regarding the application to renew the operating license for Arkansas Nuclear One, Unit 1 (ANO-1), which was filed by Entergy Operations, Inc., by letter dated January 31, 2000. The Office of Nuclear Reactor Regulation has reviewed the ANO-1 license renewal application for compliance with the requirements of Title 10 of the Code of Federal Regulations, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document its findings.

In its submittal of January 31, 2000, the Entergy Operations, Inc., requested renewal of the ANO-1 operating license (License No. DPR-51), which was issued under Section 104 of the Atomic Energy Act of 1954, as amended, for a period of 20 years beyond the current license expiration date of May 20, 2014. ANO-1 is located in Pope County in the central region of Arkansas on the shore of Lake Dardanelle. ANO-1 is a Babcock and Wilcox pressurized-water reactor nuclear steam supply system that is designed to generate 2568 MW thermal, or approximately 836 MW electric.

The NRC ANO-1 license renewal project manager is Robert J. Prato. Mr. Prato may be contacted by calling (301) 415-1147 or by writing to the License Renewal and Standardization Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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